NUREG-0797
Supplement No. 8

Safety Evaluation Report
related to the operation of Comanche Peak Steam Electric Station, Units 1 and 2

Docket Nos. 50-445 and 50-446

Texas Utilities Generating Company, et al.

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

February 1985

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ABSTRACT

Supplement 8 to the Safety Evaluation Report for the Texas Utilities Electric Company application for a license to operate Comanche Peak Steam Electric Station, Units 1 and 2 (Docket Nos. 50-445, 50-446), located in Somervell County, Texas, has been jointly prepared by the Office of Nuclear Reactor Regulation and the Comanche Peak Technical Review Team of the U. S. Nuclear Regulatory Commission. This Supplement provides the results of the staff's evaluation and resolution of approximately 80 technical concerns and allegations relating to civil and structural and miscellaneous issues regarding construction and plant readiness testing practices at the Comanche Peak facility. Issues raised during recent Atomic Safety and Licensing Board hearings will be dealt with in future supplements to the Safety Evaluation Report.

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ACRONYMS AND ABBREVIATIONS

AA	-	independent assessment program allegation
AB	-	American Bridge
AB	-	bolt allegation
ABRR	-	as-built reverification records
A-C	-	Allis-Chalmers
AC	-	concrete/rebar allegation
ACI	-	American Concrete Institute
AD	-	design of pipe/pipe support allegation
ADS	-	audit discrepancy report
AE	-	electrical allegation
AEOD	-	Office for Analysis and Evaluation of Operational Data (NRC)
AFW	-	auxiliary feedwater system
AH	-	hanger allegation
AI	-	intimidation allegation
AISC	-	American Institute of Steel Construction
ALARA	-	as low as reasonably achievable
AM	-	miscellaneous allegation
ANI	-	authorized nuclear inspector
ANS	-	American Nuclear Society
ANSI	-	American National Standards Institute
AO	-	protective coating allegation
AP	-	pipe and pipe support allegation
APC	-	AMP Product Corporation
AQ	-	quality assurance/quality control allegation
AQB	-	QA/QC bolt allegation
AQC	-	QA/QC concrete/rebar allegation
AQE	-	QA/QC electrical allegation
AQH	-	QA/QC hanger allegation
AQL	-	acceptable quality level
AQO	-	QA/QC coating allegation
AQP	-	QA/QC pipe and pipe support allegation
AQW	-	QA/QC welding allegation
ARMS	-	Automated Records Management System
ASLB	-	Atomic Safety and Licensing Board
ASME	-	American Society of Mechanical Engineers
ASTM	-	American Society for Testing and Materials
AT	-	acceptance test
AT	-	test program allegation
AV	-	vendor/generic allegation
AW	-	welding allegation
B&PVC	-	Boiler & Pressure Vessel Code
B&R	-	Brown & Root, Inc.
BNL	-	Brookhaven National Laboratory
BRHL	-	Brown & Root Hanger Locations
BRIR	-	Brown & Root Inspection Report

BRP - Brown & Root piping isometric drawing
BTP - Backfit Test Program
BWR - boiling water reactor

C&L - Corner and Lada (computer program)
C&S - civil and structural
CAR - Corrective Action Request
CASE - Citizens Association for Sound Energy
CAT - Construction Appraisal Team (NRC)
CB&I - Chicago Bridge & Iron Company
CCL - Corporate Consulting and Development Company, Limited
CCS - Component Cooling System
CCW - component cooling water
CEL - Coating Exempt Log
CFR - Code of Federal Regulations
CHN - construction hold notice
CILRT - containment integrated leak rate test
CMC - component modification cards
CMTR - certified material test report
COT - construction operation traveler
CP - Comanche Peak
CP - construction permit
CPPE - Comanche Peak Project Engineering
CPSES - Comanche Peak Steam Electric Station
CPSIG - Comanche Peak Seismic Interaction Group
CSTS - Construction and Startup/Turnover Surveillance Group (TUEC)
CVCS - chemical and volume control system
CZ-11 - Carboline Carbo zinc 11

DBA - design basis accident
DCA - design change authorization
DCC - Document Control Center (TUEC)
DCTG - Design Change Tracking Group
DCVG - Design Change Verification Group
DE - Division of Engineering (NRC)
DFT - dry film thickness
DL - Division of Licensing (NRC)
D-6 - Ameron Dimetcote 6

E&I - Electrical and Instrumentation
ECCS - emergency core cooling system
EDO - Executive Director for Operations (NRC)
ERG - emergency response guideline
ETG - Electrical Test Group (TUEC)

FDSG - Field Damage Study Group (TUEC)
FJO - field job orders
FP - fire protection
FSAR - Final Safety Analysis Report
FW - field weld

G&H - Gibbs & Hill
GAP - Government Accountability Project
GDC - general design criteria
GE - General Electric Corporation
GED - General Equivalency Diploma
GHH - Gibbs & Hill hanger (isometric drawing)

HFT - hot functional test
HIR - hanger inspection report
HP - hanger package
HP - high pressure
HVAC - heating, ventilation and air conditioning system
HX - heat exchangers

IAP - Independent Assessment Program
ICC - inadequate core cooling
IE - Office of Inspection and Enforcement (NRC)
IEB - Inspection and Enforcement Bulletin
IEEE - Institute of Electrical and Electronics Engineers
IM - interoffice memorandum (TUEC)
INPO - Institute for Nuclear Power Operations
IOM - interoffice memorandum
IR - inspection report (NRC)
IRN - item removal notice
ITT-G - ITT Grinnell

JTG - Joint Test Group (TUEC)
JUMA - Joint Utility Management Assessment Group

LE - left end
LOCA - loss of coolant accident
LP - liquid penetrant

M&P - mechanical and piping
MAR - maintenance action request
MCC - motor control center (GE)
MDB - master data base
MIFI - mechanical fabrication inspector
MIL - material identification list (or log)
MIME - Mechanical Equipment Inspector
MQE - Mechanical Quality Engineering
MR - material requisition
MRS - manufacturer's record sheet
MS - main steam (line)
MWDC - multiple weld data card

N/A - not applicable
NCR - nonconformance report (TUEC)

NDE - nondestructive examination
NDT - nondestructive testing
NI - never incorporated
NONSAT - nonsatisfactory
NOV - Notice of Violation (NRC)
NPSH - net positive suction head
NPSI - Nuclear Power Service Incorporated
NRC - U.S. Nuclear Regulatory Commission
NRR - Office of Nuclear Reactor Regulation (NRC)
NSSS - nuclear steam supply system

O&M - Operations and Maintenance (TUEC)
OBE - operating basis earthquake
OI - Office of Investigations (NRC)
OJT - on-the-job training
OL - operating license
ORNL - Oak Ridge National Laboratory

PC - protective coating
PCR - plant change request
PET - permanent equipment transfer
PFG - Paper Flow Group
PFS - pipe fabrication shop
PORV - power operated relief valve
PPM - parts per million
PSAR - Preliminary Safety Analysis Report
PSE - Pipe Support Engineering (TUEC)
PT - preoperational test
PTS - pressurized thermal shock
PWR - pipe whip restraints
PWR - pressurized water reactor
P-305 - Carboline Phenoline 305

QA - quality assurance
QAI - quality assurance investigation (TUEC)
QC - quality control
QE - quality engineer

RCB - Reactor Containment Building
RE - right end
RES - Office of Nuclear Regulatory Research (NRC)
RFIC - request for information or clarification (B&R)
RG - Regulatory Guide (NRC)
RHRS - residual heat removal system
RI - NRC Region I Office
RIR - receipt inspection report (TUEC)
RIV - NRC Region IV Office
RPE - radiation protection engineer
RPI - rod position indication
RPS - radiation protection supervisor

RPS - report process sheet (TUGCO)
RPV - reactor pressure vessel
RPVRI - reactor pressure vessel reflective insulation
RRI - Resident Reactor Inspector (NRC)
RV - reactor vessel
RWN - room work notifications

SAP - startup administration procedure
SALP - Systematic Assessment of Licensee Performance (NRC)
SAT - satisfactory
SAVC - structural assembly verification card
SER - Safety Evaluation Report (NRC)
SI - safety injection
SIS - Special Inspection Services
SMAW - shielded metal arc welding
SNM - special nuclear material
SORC - Station Operations Review Committee
SRIC - Senior Resident Inspector for Construction (NRC)
SRP - Standard Review Plan (NRC)
SRT - Special Review Team (NRC)
SSE - safe shutdown earthquake
SSER - Safety Evaluation Report Supplement
SSI - safe shutdown impoundment
SSPC - Steel Structures Painting Council
SSWP - station service water pumps
STE - system test engineer
SWA - startup work authorization
SWO - shop work order

TDCR - test deficiency change request
TDI - Transamerica Delaval, Inc.
TDR - test deficiency report
10 CFR 50 - Title 10 Code of Federal Regulations Part 50
TI - temporary instruction
TIDC - Division of Technical Information and Document Control (NRC)
TNE - TUEC Nuclear Engineering
TP - test program
TPD - test procedure deviation
Tr - transcript
TRT - Technical Review Team (NRC)
TSABC - technical services as-built coordinator
TSDR - technical services design review coordinator
TSI - thermolag
TSMD - Technical Services Mechanical Drafting
TSP - tri-sodium phosphate
TUEC - Texas Utilities Electric Company
TUGCO - Texas Utilities Generating Company
TUSI - Texas Utilities Service, Inc.

UCC - University Computing Company
USI - unresolved safety issue
UT - ultrasonic test
UTA - University of Texas at Austin

VCD - vendor-certified drawing
VT - visual weld (inspector)

W - Westinghouse Electric Corporation
WDC - weld data card
WFML - weld filler metal log
WPS - welding procedure specification

1 INTRODUCTION

On July 14, 1981, the U.S. Nuclear Regulatory Commission (NRC) issued a Safety Evaluation Report (SER) (NUREG-0797) related to the application by the Texas Utilities Electric Company (TUEC) for a license to operate Comanche Peak Steam Electric Station (CPSES) Units 1 and 2. Subsequently seven supplemental Safety Evaluation Reports (SSERs) were issued by the staff. Supplement No. 7, published in January 1985, dealt with technical concerns and allegations in the electrical and instrumentation and test program areas about Comanche Peak. This report, Supplement No. 8, is the second of a series of SSERs dealing with various technical concerns and allegations about Comanche Peak. This report addresses approximately 80 technical concerns and allegations relating to civil and structural and miscellaneous issues. Appendix K to this report provides details of the staff's evaluation and findings of these technical concerns and allegations.

The technical concerns and allegations about Comanche Peak were part of the regulatory issues that remained outstanding toward the completion of construction of the Comanche Peak facility. The NRC's Executive Director for Operations (EDO) issued a directive on March 12, 1984, establishing a program for assuring the overall coordination/integration of these issues and their resolution prior to the staff's licensing decision. In response to the EDO's directive, a program plan was developed and approved on June 5, 1984, by the Directors of NRC's Office of Inspection and Enforcement, Office of Nuclear Reactor Regulation, and the Administrator of NRC's Region IV Office. This program plan, entitled Comanche Peak Plan for the Completion of Outstanding Regulatory Actions, specified the critical path issues, addressed the scope of work needed, and provided a project schedule for completion.

Management and coordination of all the outstanding regulatory actions for Comanche Peak are under the overall direction of Mr. Vincent S. Noonan, the NRC Comanche Peak Project Director. Mr. Noonan may be contacted by calling 301-492-7903 or by writing to the following address:

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Copies of this Supplement are available for public inspection at the NRC's Public Document Room at 1717 H Street, NW, Washington, D.C. 20555, and the Local Public Document Room, located at the Somervell County Public Library On the Square, P.O. Box 1417, Glen Rose, Texas, 76043. Availability of all material cited is described on the inside front cover of this report.

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APPENDIX K

STATUS OF STAFF EVALUATION

AND RESOLUTION OF TECHNICAL CONCERNS

AND ALLEGATIONS RELATING TO

CIVIL AND STRUCTURAL AND MISCELLANEOUS ISSUES

REGARDING CONSTRUCTION AND PLANT

READINESS TESTING AT

COMANCHE PEAK STEAM ELECTRIC STATION

UNITS 1 AND 2

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1. Introduction

As construction of the Comanche Peak Steam Electric Station was nearing completion, issues that remained to be resolved prior to the consideration of issuance of an operating license were complex, resource intensive, and spanned more than one NRC office. To ensure the overall coordination and integration of these issues, and to ensure their resolution prior to licensing decisions, the NRC's Executive Director for Operations (EDO) issued a memorandum on March 12, 1984, directing the NRC's Office of Nuclear Reactor Regulation to manage all necessary NRC actions leading to prompt licensing decisions, and assigning the Director, NRC's Division of Licensing, the lead responsibility for coordinating and integrating the related efforts of various offices within the NRC.

The principal areas needing resolution before a licensing decision on Comanche Peak can be reached include: (1) the completion and documentation of the staff's review of the Final Safety Analysis Report (FSAR); (2) those issues in contention before the NRC's Atomic Safety and Licensing Board (ASLB); (3) the completion of necessary NRC regional inspection actions; and (4) the completion and documentation of the staff's review of technical concerns and allegations regarding design and construction of the plant.

Technical concerns and allegations about Comanche Peak, totalling approximately 900, have been raised mainly by the quality assurance/quality control (QA/QC) personnel working or having worked on site. Their job responsibilities involve or involved QA/QC aspects of safety-related structures, systems, and components to determine whether and to what extent such items are manufactured, purchased, stored, maintained, installed, tested, and inspected as required by project documents and procedures. Many of these allegations were made orally to NRC Region IV staff, NRC Comanche Peak Site Resident Inspectors, NRC investigators, or in letters to the NRC, as well as in testimony before the Atomic Safety and Licensing Board (ASLB). Individuals with allegations were also sponsored by the intervenor group Citizens Association for Sound Energy (CASE) and the Government Accountability Project (GAP). General allegations about poor construction work at Comanche Peak were also made in several newspaper articles in the Dallas/Fort Worth, Texas areas.

By the end of April 1984, the staff identified approximately 400 technical concerns and allegations related to the construction of the Comanche Peak facility, including findings by NRC's Special Review Team. (See Section 2.1 below.) During its investigation of a concern or allegation, the TRT identified additional concerns. Interviews with allegers also yielded additional concerns. By December 1984, approximately 600 concerns and allegations had been identified. In addition, approximately 300 allegations were recently provided to the TRT by one alleger.

These technical concerns and allegations were grouped by subject into the following areas:

- Electrical and Instrumentation
- Civil and Structural

- Mechanical and Piping
- Quality Assurance and Quality Control (QA/QC)
- Coatings
- Test Program
- Miscellaneous

This report is the second of a series of reports dealing exclusively with the NRC staff's efforts to evaluate and resolve the technical concerns and allegations raised by various parties and individuals regarding construction practices at the Comanche Peak facility. Included in this report are civil and structural and miscellaneous issues. An allegation or concern was assessed as having no safety significance if, based on technical findings, the assessment showed that a structure, component, or system would perform its intended function. Subject areas covered in this report include civil and structural and miscellaneous issues. A report on the electrical and instrumentation and test program areas was published in January 1985. The technical concerns and allegations in the areas of mechanical and piping, coatings, and QA/QC, as well as the remaining areas of outstanding regulatory actions, will be addressed in future supplements to the Comanche Peak Safety Evaluation Report (SER).

The staff's findings for civil and structural and miscellaneous allegations or concerns are summarized in Section 3 of this Appendix. Attachment 1 to the Appendix is a listing of the technical concerns and allegations relating to civil and structural and miscellaneous issues. Details of the assessment and findings on individual concerns or allegations appear in Attachment 2 to this Appendix. Those aspects of the concerns or allegations that pertain to wrongdoing (e.g., falsification of records) were forwarded to the NRC's Office of Investigations (OI) for followup because they are outside the scope of the technical staff's review.

A number of potential violations of NRC rules and regulations have been identified during the course of the TRT investigation. These potential violations have not been addressed in this SSER, but will be further reviewed by the NRC Region IV staff, which will determine appropriate followup actions.

2. Comanche Peak Technical Concerns and Allegations Management Program

2.1 Background

Shortly after the EDO's issuance of the March 12, 1984, directive, the staff found it necessary to (1) obtain current information relative to TUEC's management control of the construction, inspection, and test program and (2) obtain necessary information to establish a management plan for resolution of all outstanding licensing actions. In order to achieve these goals in an expeditious and objective manner, a Special Review Team (SRT) was formed to conduct an unannounced review of the Comanche Peak plant. The SRT consisted of eight reviewers and one team leader, all from NRC's Region II Office, and a team manager from NRC headquarters. The SRT spent over 800 manhours, from April 3 to April 13, 1984, performing this review. The SRT concluded that TUEC's programs were being sufficiently controlled to allow continued plant construction while the NRC completed its review and inspection of the Comanche Peak facility.

The SRT review also provided a basis for the development of an NRC management plan for the resolution of all outstanding licensing actions. This plan was approved on June 5, 1984, by the Directors of NRC's Office of Inspection and Enforcement, Office of Nuclear Reactor Regulation, and the Administrator of NRC's Region IV Office. The purpose of the plan was to ensure the overall coordination and integration of the outstanding regulatory actions at Comanche Peak and their satisfactory resolution prior to a licensing decision by the NRC. In accordance with the plan, a Technical Review Team (TRT) was formed to evaluate and resolve technical issues and those allegations that had been identified. On July 9, 1984, the TRT began its 10-week (five 2-week sessions) onsite effort, including interviews of allegers and TUEC personnel, to determine the validity of the technical concerns and allegations, to evaluate their safety significance, and to assess their generic implications. The TRT consisted of about 50 technical specialists from NRC headquarters, NRC Regional Offices, and NRC consultants, who were divided into groups according to technical discipline. Each group was also assigned a group leader.

2.2 Review Approach and Methodology

2.2.1 Concern and Allegation Tracking System

A tracking system was developed for identifying and listing each concern or allegation. These technical concerns and allegations were grouped according to their topical areas or disciplines, and were listed numerically within each group in the order that they were identified by the TRT. The tracking system included a description of the concern or allegation; its status or the actions taken to resolve it; the nature of the sources of the concern or allegation (i.e., anonymous or confidential); a code for the individual who identified the concern or allegation (instead of the individual's name); the date when the concern or allegation was received by the TRT; the source document (e.g., letter, NRC inspection report, hearing transcript, etc.); cross reference; etc. At the end of each 2-week session, the concern/allegation tracking system was updated, as needed, to reflect the status of each concern or allegation, as well as any new ones that had been added.

2.2.2 Review Methodology

The technical concerns or allegations similar in subject were combined and evaluated as one category. For each concern/allegation or concern/allegation category, an approach to resolution was developed by the cognizant reviewer(s). Each approach to resolution was reviewed and approved by the responsible group leader. The group leaders and reviewers were instructed to:

- develop and maintain a work package for each issue or category of issues that contained or referenced pertinent documentation associated with the issue(s) and the ultimate resolution, including records of interviews and inspections for supporting the final NRC staff decisions regarding the issue(s); and to

- protect the identity of the allegers, as a matter of NRC practice. Such efforts included limited and controlled distribution of allegation-related documentation and correspondence; minimal use of names, identifying titles, or position descriptions in written material; enlarged sampling of activities to prevent direct links by non-NRC personnel between the activity under investigation and the alleger; and other indirect approaches toward investigating the allegations.

During TRT onsite sessions, daily meetings were held at the review group level to assess progress, to adjust the inspection and evaluation approach as needed, and to provide a forum for the reviewers to interact with one another or to discuss problems and to arrive jointly at resolutions. Similar daily meetings were also held at the management level where the group leaders interacted with one another and with the Project Director, his assistant and staff.

In evaluating the technical concerns and allegations, the TRT reviewers examined areas in the plant where direct observation could provide information needed for evaluating an allegation or concern. During its onsite sessions, the TRT interviewed the allegers as needed to clarify their concerns or allegations. To the extent possible, the TRT contacted allegers after its onsite review to discuss preliminary TRT findings and to obtain any additional comments from them. (See Section 2.2.3 below.) The TRT also interviewed TUEC and TUEC contractor personnel as was warranted by the evaluation. In addition to these contacts, the TRT reviewed various project documents, including specifications, engineering drawings and analyses, procedures, instructions, NRC Region IV inspection reports, and applicable sections of the Final Safety Analysis Report (FSAR) and NRC regulations pertinent to the allegation or sample selected by the TRT for inspection. The TRT also examined construction records, such as design change authorizations, construction work packages, QC inspection reports, nonconformance reports, deficiency logs, lists and reports, and QC inspector training and certification records. In addition, the TRT reviewed pertinent transcripts from recent ASLB hearings and depositions of TUEC personnel and former employees.

Based on these reviews and interviews, the TRT determined the validity of each technical concern or allegation and assessed its safety significance, its potential generic implications, and any indications of potential management breakdown. Detailed documentation of the TRT assessment and final determinations of each technical concern or allegation appear in Attachment 2 to this Appendix.

2.2.3 Interviews with Allegers

Approximately 900 technical concerns and allegations regarding the construction of the Comanche Peak facility have been raised by approximately 70 allegers through various mechanisms. During its onsite work, the TRT interviewed 18 individuals in person, some of whom received followup interviews by telephone. For ten allegers, the TRT reviewers were able to obtain the needed information by telephone and determined that personal interviews would not be necessary. Three allegers contacted by the TRT declined being interviewed. Five allegers could not be located during the TRT's onsite sessions because their current addresses and telephone numbers were not available. They have not responded to correspondence from the TRT sent to their last known addresses expressing the TRT's intention to discuss their concerns with them. Efforts to locate these individuals included inquiries through the NRC's Office of Investigations, NRC's Region IV staff, the telephone company and U.S. Postal Service, selected inquiries of their relatives and former co-workers, confidential examination of the personnel files of TUEC and its contractors, and in some cases, inquiries to the intervenor group, the Citizens Association for Sound Energy (CASE), and the Government Accountability Project (GAP).

To the extent possible, the TRT kept a transcript for each personal interview conducted during its onsite sessions. The names and identities of the allegers had been deleted from the transcripts, as well as from other pertinent reference or source documents, before TRT reviewers were given any portions of these documents for review and follow-up. During the TRT's onsite work, the original transcripts were kept in a locked file in the TRT Project Director's office. The distribution of these transcripts within the NRC, and even within the TRT, was limited and controlled.

Subsequent to its onsite work, and at the completion of its evaluation, the TRT attempted to contact each alleger to discuss the TRT's findings regarding their original concerns, and to obtain additional comments from them, if any. Thirty allegers have received such followup interviews. A total of 19 allegers could not be located. Some of these individuals had received initial TRT interviews but had since left the area. Three allegers declined to have further contacts with the TRT. The TRT is in the process of contacting the remaining allegers for followup interviews. The outcome of followup interviews conducted through December 1984, is briefly discussed in the individual SSER sections in Attachment 2. Transcripts were kept for all followup interviews conducted either by telephone or in person.

2.3 Communications with TUEC

Whenever the TRT reviewers encountered problems during their evaluations, the TRT Project Director and/or his designee resolved them through discussions with TUEC management onsite. There were also frequent staff-level contacts between TRT members and TUEC personnel during the TRT's onsite activities. In keeping with the NRC practice of promptly notifying applicants of outstanding information/evaluation needs that could potentially affect plant safety, the staff held several meetings with TUEC representatives at NRC headquarters toward the end of the TRT's review. These meetings were held to discuss potential safety concerns and to request additional information needed by the TRT to complete its review.

The NRC staff met with TUEC representatives for the first of these meetings on September 18, 1984, to discuss TRT findings for electrical and instrumentation, civil and structural, and test program allegations and concerns. A letter documenting these findings and a request for additional information was issued to TUEC on the day of the meeting (Attachments 3 and 4). TUEC later submitted the requested information in the form of a proposed program plan, delineating planned actions to address the deficiencies identified by the TRT. The TRT met with TUEC representatives to discuss this proposed program plan on October 19 and 23, 1984. TUEC submitted a partially revised program plan to NRC on November 21, 1984. By letter dated January 24, 1985, the TRT provided TUEC with detailed comments on the program plan and issue-specific action plans. On November 29, 1984, NRC sent a letter to TUEC containing potential open issues and requesting additional information and proposed program plans for mechanical and piping and miscellaneous allegations and concerns (Attachment 5). The letter also provided TUEC with the status of NRC's evaluation of coatings allegations. Informal telephone discussions between TRT group leaders and their TUEC counterparts regarding these letters have been ongoing. (Reports documenting these discussions have been made available to CASE and are available for inspection at the NRC Public Document Room, 1717 H St., N.W., Washington, D.C. 20555, and at the Comanche Peak Local Public Document Room, Somervell County Public Library On The Square, P.O. Box 1417, Glen Rose, Texas 76043.) On January 8, 1985, the NRC issued a letter to TUEC informing them of the TRT's preliminary findings in the construction QA/QC area and requesting a program and schedule for completing a detailed and thorough assessment of the QA issues presented in the letter. A meeting between TUEC and the TRT was held on January 17, 1985, to discuss potential open issues in the QA/QC area. TUEC's proposed program plan for each of the subject areas and its implementation of the plan will be evaluated by the NRC staff prior to the NRC licensing decision on Comanche Peak.

3. Summary of Evaluations

3.1 Civil and Structural (C&S) Group Summary

3.1.1 Scope of Concerns and Allegations

The concerns and allegations in the C&S discipline involved most aspects of reinforced concrete construction and testing. These allegations and concerns relate to (1) design deficiencies, (2) testing or inspection irregularities, (3) incorrect construction practices, (4) inadequate repairs, (5) uncorrected, unsafe conditions in the completed structures, and (6) premature structural loading. The total of 57 concerns and allegations were grouped by subject into the following 17 categories:

Category No.	Subject	Characterization of Concerns and Allegations
1	Inadequate Materials Used in Concrete	Rejected aggregate was incorporated in the basemat of Unit 1 reactor; unauthorized quantities of water added to concrete used in basemat; rejected concrete placed in turbine generator building; concrete with excessive slump placed in containment walls; concrete rejected for being over specification limit on time to discharge was placed in the Circulating Water Intake Structure.
2	"Bad Concrete Work" and "Sloppy" Placement of Concrete	"Bad concrete work" and "sloppy" placement of concrete; placement of "soupy" concrete in a slab in the Auxiliary Building in the summer of 1976.
3	Placement of Concrete During Poor Weather Conditions	Placement of concrete during rainstorm and without approval by QC personnel and during or immediately before freezing weather; some field-cured cylinders and standard-cured cylinders failed specification requirements for concrete strength, and the Schmidt rebound hammer test was misapplied.

Category No.	Subject	Characterization of Concerns and Allegations
4	Concrete Void, Cracks, and Crumbling	Voids in concrete behind stainless steel liner of Unit 1 reactor cavity and in building walls; cracks in concrete basemat of Unit 1 and in floor slabs in the plant building; foreign material embedded in concrete; fresh concrete placed on top of crumbling concrete.
5	Miscellaneous Concrete Construction Irregularities	Equipment was set on grout before the grout properly gained strength through aging; hanger inserts installed at improper angles; trash in bottom of a form was covered with concrete.
6	Rebar Improperly Installed or Omitted	Rebar was installed that was not properly inspected upon receipt at the site; rebar omitted at various specified locations.
7	Uncontrolled Repair of Concrete	A hole in a concrete slab resulting from removal of a Hilti bolt in the floor of the Safeguards Building was repaired in an "uncontrolled manner."
8	Falsification of Records	Various specified records concerning concrete tests were falsified.
9	Improperly Conducted Inspector Recertification Tests	Inspector recertification tests were done "open book" after March of 1977 and examinations were given with answers provided.
10	Violations of Testing Procedures	Equipment required for aggregate testing unused; short cuts taken in aggregate testing; concrete placed without required testing; concrete cylinder compression tests run at faster loading rate than permitted by NRC regulations; concrete test cylinders in the laboratory moist room allowed to dry.

Category No.	Subject	Characterization of Concerns and Allegations
11	Poor Workmanship in Use of Rotofoam	Poor workmanship in use of rotofoam as a temporary spacer during construction to maintain required seismic gap between Category I concrete structures.
12	Concrete Construction Deficiencies	A spillway pillar, span, or column was erected 75 to 80 degrees offset.
13	Concrete Cracks At Bottom of Reactor Vessel	Detrimental cracks in concrete pad at bottom of reactor vessel.
14	Control Room Area Deficiencies	The field run conduit, drywall, and lighting fixtures installed above ceiling panels in the control room are classified as nonseismic and are supported only by wires, and may fall as a result of a seismic event.
15	Unauthorized Cutting of Rebar	Undocumented and unauthorized holes were drilled through rebar.
16	Excavation Overbreak/ Seismic Response	Overexcavation and improper fill under Unit 1 Containment Building could invalidate expected seismic response of the foundation due to change in properties resulting from removal of in-situ material.
17	Improper Concrete Sampling	Personnel produced incorrect readings on concrete batch plant scales by leaning on wires connecting the weighing hoppers to the scales.

3.1.2 Civil and Structural (C&S) Group

The Civil and Structural Group consisted of three NRC employees and four consultants, all of whom are civil and structural engineers, with a combined total of 137 years of experience in general design and in nuclear and non-nuclear heavy construction work.

These reviewers were selected for their technical expertise and experience in design, construction, quality assurance, and ability to detect discrepancies in construction records.

3.1.3 Findings for Civil and Structural Issues

Fourteen of the 57 concerns and allegations reviewed by the TRT in the Civil and Structural area were not substantiated. Of the 20 that were substantiated, 3 were found to have potential safety significance. In addition, there are 2 allegations, although not substantiated, whose safety significance can not be determined at this point. TUEC has been requested to provide more information to the NRC staff before these issues can be resolved. The TRT could not determine the validity of 21 allegations. However, a conservative approach was taken to disposition the allegation, i.e., the TRT assessed the potential structural significance of the allegation assuming that it was true. Two allegations simply reiterated allegations already made.

The first issue that has been substantiated and was determined to be of potential safety significance involved reinforcing steel (rebar) omitted from a concrete placement in the reactor cavity wall of Unit 1. The C&S Group requested documentation indicating that an analysis was performed supporting the omission of this rebar. The C&S group was subsequently informed that an analysis had not been performed. Therefore, the safety significance of this issue cannot be determined until an analysis is performed verifying the adequacy of the reinforcing steel as installed. (See Attachment 2, C&S Category 6.)

Another issue that has been substantiated and was determined to be of potential safety significance concerned the maintenance of an air gap between concrete structures. Based on a review of available inspection reports and related documents, on field observations, and on discussions with TUEC engineers, the C&S Group could not determine if an adequate air gap had been provided between concrete structures. In addition, it is not apparent that the permanent installation of elastic joint filler material (rotofoam) between the Safeguards Building and the Reactor Building, and below grade for the other concrete structures, is consistent with the assumptions and dynamic models used to analyze the buildings. (See Attachment 2, C&S Category 11.)

The third issue that has been substantiated and was determined to be of potential safety significance concerned the seismic design of the control room ceiling elements. This issue was jointly reviewed by the C&S Group and the Electrical and Instrumentation Group. For the nonseismic items (other than the sloping suspended drywall ceiling), and for nonsafety-related conduits whose diameter is 2 inches or less, the C&S Group could find no evidence that the possible effects of a failure of these items had been considered. In addition, the C&S Group determined that calculations for seismic Category II components (e.g., lighting fixtures) and the calculations for the sloping suspended drywall ceiling did not adequately reflect the rotational interaction with the nonseismic items. The fundamental frequencies of the supported masses had not been determined to assess the influence of the seismic response spectrum at the control room ceiling elevation on the seismic response of the ceiling elements. (See Attachment 2, C&S Category 14.)

The C&S group investigated the technical implications concerning falsification of concrete strength tests. The preponderance of evidence suggests that falsification did not occur. However, since a number of other allegations were resolved on the basis of concrete strength results, the C&S Group believes that action is required on the part of TUEC to provide confirmatory evidence that the reported concrete strength test results are indeed representative of the strength of the concrete placed in Category I concrete structures. (See Attachment 2, C&S Category 8.)

Another issue that was not substantiated and whose safety significance could not yet be determined concerned the unauthorized cutting of rebar in the Fuel Handling Building. The C&S Group found that if certain holes were drilled to the depth alleged, rebar would have been cut without authorization. (See Attachment 2, C&S Category 15.)

The C&S Group found that the allegation concerning hollow places in concrete behind the stainless steel liner of the Unit 2 Reactor Cavity is true; the hollow places are currently undergoing repairs. The repairs and the repair documentation must be inspected, reviewed, and approved by the NRC before the TRT can determine that this issue has been adequately resolved. (See Attachment 2, C&S Category 4.)

The C&S Group could not substantiate the concerns raised by the remaining allegations and concluded that these concerns have no structural safety significance. However, the results of the evaluations for Categories 1, 3, 4, 5, 6, 7, 8, 9, 10, 15, and 17 are being further assessed by the QA/QC Group as part of its overall programmatic review. (See Attachment 2, C&S Categories 1, 3, 4, 5, 6, 7, 8, 9, 10, 15 and 17.)

3.1.4 Overall Assessment and Conclusions

During its evaluations, the TRT reviewed pertinent construction records (e.g., concrete placement packages, NCRs, concrete test results), structural design drawings and calculations, specifications (e.g., for concrete reinforcing steel), interviewed craft and TUEC personnel and conducted plant inspections. This documentation, to the extent reviewed by the TRT, was judged to be adequate and consistent with applicable FSAR commitments, except for the deficiencies identified in the SSER sections in Attachment 2. Therefore, the TRT concludes that the civil and structural construction within the scope of the TRT C&S group review effort was adequate and was, for the most part, well documented.

Five issues in the civil and structural area still require further action. One case involving reinforcing steel omitted from the reactor cavity wall, and another case of alleged unauthorized drilling of reinforcing steel, require further documentation. TUEC must also test concrete in place to evaluate an allegation concerning falsified concrete strength tests. In addition, TUEC must conduct analyses and inspections to determine whether the separation between buildings is adequate to provide acceptable performance in an earthquake. Finally, there must be a seismic analysis of the suspended ceiling, lighting fixture and nonsafety-related conduit in the control room to demonstrate design adequacy of the ceiling elements. The potential safety implications of this issue for nonseismic structures, systems, and components in other parts of the plant must also be evaluated.

3.2 Miscellaneous Group Summary

3.2.1 Scope of Concerns and Allegations

The allegations with a Miscellaneous designation covered a wide variety of topics and involved both administrative and construction activities, some safety related and some nonsafety related. In total, 29 allegations were designated as Miscellaneous; i.e., their subject matter did not fall within the scope of responsibility of one of the other Technical Review Team's technical disciplines. Twenty-five allegations were subsequently consolidated into 20 categories, each of which dealt with a general topic; three were transferred to the TRT Mechanical and Piping group for review and followup and one to the Office of Investigations. The following is a listing and description of each of the 20 Miscellaneous categories.

Category No.	Subject	Characterization of Concerns and Allegations
1	Nuclear Fuel	Nuclear fuel was received prior to issuance of special nuclear material license.
2	Reactor Pressure Vessel (RPV)	Expansion of the RPV during hot functional testing caused the vessel reflective insulation to come in contact with the concrete biological shield wall; the Unit 1 RPV is located 3/16 inch off center.
3	Comanche Peak PSAR	The Comanche Peak PSAR contains errors.
4	Radioactive Material thrown into Comanche Peak Reservoir	Radioactive material was thrown into the lake.
5	High Pressure Turbine	Cracks were observed in lower casing of the high pressure turbine.
6	Pressurizer Area Piping	A section was cut from a prefabricated pipe "in the pressurizer area."
7	Unit 1 Main Condenser	Design and fabrication problems were associated with the main condenser.
8	Component Cooling Water Surge Tank	Anchor bolts were damaged during installation.
9	Hayward Tyler Pump Deficiencies	Hayward Tyler pumps in safety systems may have unidentified deficiencies because of a poor quality assurance program at Hayward Tyler.
10	Unit 1 Diesel Generators	Two Unit 1 diesel generators were damaged.
11	Polar Crane	Shimming and installation of the polar crane were improper.
12	Missile Barrier Door	A deficient weld on a door was accepted.

Category No.	Subject	Characterization of Concerns and Allegations
13	Tube to Base Plate Weldments	Tube steel was cut at the wrong angle and welded to a baseplate, leaving a large gap between the tube and baseplate.
14	NRC Form-3 Posting	NRC Form-3 was posted at an insufficient number of site locations.
15	Drug Abuse	Drug use and abuse was widespread and management did not give proper attention to the alleged problem.
16	HVAC	Heating, ventilating, and air conditioning system (HVAC) supports for seismic loads were not analyzed; HVAC components and supports inside containment were not properly considered as missiles; HVAC failure during a postulated accident would allow temperatures to rise to an unacceptable level inside containment.
17	Reactor Vessel Internals	Damage occurred to upper internals of the reactor vessel.
18	Polar Crane Cables	Internal wires were broken in the polar crane festooned cables.
19	Radwaste System Contamination	Workers habitually urinated on stainless steel pipe.
20	Instructions to Craft Personnel	Inadequate rigging and handling instructions were provided to craft personnel.

3.2.2 Miscellaneous Group

The members of the Miscellaneous Group were assembled based on their technical expertise, capabilities, and experience in engineering design, quality assurance and document control, inspection, construction, and regulatory activities. The group included five members from NRC's Region IV office, with expertise in various technical disciplines, and three consultants. Collectively, the group possessed experience in excess of 50 years in the nuclear power industry and its regulation.

3.2.3 Findings for Miscellaneous Issues

Fourteen of the 24 allegations (grouped into 20 categories) pertained to systems and components classified as nonnuclear safety (NNS) in the Comanche Peak Steam Electric Station (CPSES) Final Safety Analysis Report (FSAR). These allegations (AM-3, 4, 5, 6, 9, 15, 16, 17, 22, 24, 25, and 30) were also not listed as quality assurance (i.e., safety-related) items in Table 17A of Volume XIV of the FSAR. Accordingly, 10 CFR Part 50 Appendix B quality assurance requirements would not apply to these systems and components except for seismic considerations. Seven of the 14 allegations have seismic classifications. Ordinarily the NRC does not inspect items that are classified NNS or nonsafety related but would observe and bring deficient non-Q items to the attention of Texas Utilities Electric Company (TUEC) for resolution. However, these items were inspected by the Technical Review Team (TRT) because the TRT was responsible for resolving all allegations and assuring that nonsafety issues did not have safety implications. Four of the 14 allegations were potentially safety significant and potentially had generic implications; however, TUEC had identified and corrected the problems concerning part of AM-25 (crane movement) and AM-30. Only AM-15 and 16 (Polar Crane Shimming) and AM-3 (Reactor Pressure Vessel Reflective Insulation) remain unresolved at this time and are identified as the first and second issues in the following paragraph.

Ten of the 24 allegations (AM-2, 7, 12, 13, 14, 18, 19, 21, 23(a) and 23(b)) pertained to matters or systems which are classified as safety related. Four of these 10 allegations (AM-13, 14, 21 and 23(a)) were potentially safety significant and had generic implications. However, TUEC had identified and corrected (or was in the process of correcting) problems described in allegations.

The first issue having potential safety significance (AM-3) involved the gap between the reactor pressure vessel insulation and the biological shield wall. Investigation of the allegation that the Unit 1 reactor pressure vessel outer wall was touching the concrete biological shield wall indicated that this allegation was not factual. However, a significant construction deficiency report documented that unacceptable cooling occurred in the annulus between the reactor pressure vessel reflective insulation (RPVRI) and the shield wall during hot functional testing, apparently because of the existence of an inadequately sized annulus gap and possibly because of the presence of construction debris in the annulus. TUEC corrected the situation by modifications to allow increased air flow for proper heat dissipation and by removal of the construction debris. TUEC representatives indicated that testing to verify the adequacy of the cooling flow will take place when additional hot functional testing is conducted. Information gathered during the investigation indicated that a design change in the RPVRI support ring (i.e., locating the ring outside rather than inside the insulation) resulted in a limited clearance between the RPVRI and the shield wall. However, TUEC failed to: (1) address the fundamental issue of the design change impact on annulus cooling flow, and (2) determine whether Unit 2 was similarly affected. Consequently, further action is required. (See Attachment 2, Miscellaneous Category 2.)

The second issue having potential safety significance (AM-15 and 16) involves the polar crane rail support system. The installation of the polar crane rail support system was investigated by visual inspection, review of associated documentation, and discussions with TUEC representatives and their contractors. Region IV documented that gaps on the Unit 1 polar crane bracket and seismic connections exceeded design requirements. In TUEC responses, the gaps were attributed to crane and bolting self-adjustment resulting from crane operation. A site design change was issued to document the acceptability of the gaps in excess of 1/16 inch which were identified in the NRC inspection report.

During further investigation of the allegation that shims for the rail support system of the polar crane had been altered during installation, gaps, which may have been excessive, were observed between the crane girder and the girder support bracket. Detailed specifications addressing the gap tolerance in the girder seat connections did not exist; however, Gibbs & Hill indicated in a November 28, 1977 letter (GHF-2207) that seated connections do not require shimming, since the area in bearing is at least the width of the bottom flange of the crane girder. Contrary to this assumption, nine girders were observed to have gaps which extended under the bottom flange that reduced the bearing surface to less than the 20-inch flange width stated in the letter. The TRT also observed conditions which indicated that the crane rail may still be moving in a circumferential direction, that three rail-to-rail ground wires were broken, that two shims have partially worked out from under the rail, and that two Cadwelds were broken. (See Attachment 2, Miscellaneous Category 11.)

The TRT found that 21 of 24 allegations (that is, all except AM-3, 15 and 16) were either unfounded or involved nonsafety-related issues, or the deficiency was identified by TUEC's quality assurance/quality control program and corrective actions had been completed that were acceptable to the TRT.

3.2.4 Overall Assessment and Conclusions

The TRT found that 9 of the 24 allegations were substantiated, were potentially safety significant, and had generic implications. However, actions taken because of NRC Bulletins, inspections, and TUEC audits/evaluations corrected all but two problems. Therefore, the TRT concludes that 21 of 24 allegations had neither safety significance nor generic implications. The two problems for which TUEC will have to complete actions and address issues are Miscellaneous Category 2, the gap between the reactor pressure vessel reflective insulation and the biological shield wall, and Miscellaneous Category 11, improper shimming and installation of the polar crane rail support system. (See Section 4.2.) Once these actions are satisfactorily completed by TUEC and are reviewed and accepted by the NRC, a finding can then be made that no outstanding issues raised by the miscellaneous concerns and allegations remain that would preclude licensing of CPSES Unit 1.

4. Actions Required of TUEC

TUEC shall submit additional information to the NRC, in writing, including a program and schedule for completing a detailed and thorough assessment of the issues identified in the following sections. This program plan and its implementation will be evaluated by the staff before NRC considers the issuance of an operating license for Comanche Peak, Unit 1. The program plan should address the root cause of each problem identified and its generic implications on safety-related systems, programs, or areas. The collective significance of these deficiencies should also be addressed. The program plan should also include the proposed TUEC action to ensure that such problems will be precluded from occurring in the future. The specific actions required of TUEC are described in the following sections.

4.1 Civil and Structural (C&S) Area

4.1.1 Rebar Improperly Installed or Omitted (See Attachment 2, C&S Category 6)

- Provide an analysis of the as-built condition of the Unit 1 reactor cavity that verifies the adequacy of the reinforcing steel between the 812-foot and 819-foot, ½-inch elevations. The analysis shall consider all required load combinations.

4.1.2 Falsification of Concrete Compression Strength Test Results (See Attachment 2, C&S Category 8)

- Determine areas where safety-related concrete was placed between January 1976 and February 1977, and provide a program to assure acceptable concrete strength. The program shall include tests such as the use of random Schmidt hammer tests on the concrete in areas where safety is critical. The program shall include a comparison of the results with the results of tests performed on concrete of the same design strength in areas where the strength of the concrete is not questioned to determine if any significant variance in strength occurs. TUEC shall submit the program for these tests to the NRC for review and approval prior to performing the tests.

4.1.3 Maintenance of Air Gap Between Concrete Structures (See Attachment 2, C&S Category 11)

- Perform an inspection of the as-built condition to confirm that adequate separation for all seismic Category I structures has been provided.

- Provide the results of analyses which demonstrate that the presence of rotofoam and other debris between all concrete structures (as determined by inspections of the as-built conditions) does not result in any significant increase in seismic response or alter the dynamic response characteristics of the Category I structures, components, and piping when compared with the results of the original analyses.

4.1.4 Seismic Design of Control Room Ceiling Elements (See Attachment 2, C&S Category 14)

- Provide the results of seismic analysis which demonstrate that the non-seismic items in the control room (other than the sloping suspended dry-wall ceiling) satisfy the provisions of Regulatory Guide 1.29 and FSAR Section 3.7B.2.8.

- Provide an evaluation of seismic design adequacy of support systems for the lighting fixtures (seismic Category II) and the suspended drywall ceiling (nonseismic item with modification) which accounts for pertinent floor response characteristics of the systems.

- Verify that those items in the control room ceiling not installed in accordance with the requirements of Regulatory Guide 1.29 satisfy applicable design requirements.

- Provide the results of an analysis that justify the adequacy of the nonsafety-related conduit support system in the control room for conduit 2 inches or less in diameter.

- Provide the results of an analysis which demonstrate that the foregoing problems are not applicable to other category II and nonseismic structures, systems, and components elsewhere in the plant.

4.1.5 Unauthorized Cutting of Rebar in the Fuel Handling Building (See Attachment 2, C&S Category 15)

- Provide information to demonstrate that only the No. 18 reinforcing steel in the first layer of the floor slab at the 810-ft, 6-inch elevation of the Fuel Handling Building was cut during installation of the trolley process aisle rails, or

- Provide design calculations to demonstrate that structural integrity is maintained if the No. 18 reinforcing steel on both the first and third layers of the floor slab was cut.

4.1.6 Hollow Places in Concrete Behind Unit 2 Reactor Cavity Liner (See Attachment 2, C&S Category 4)

- Provide details of the successful completion of the repairs to the hollow places in concrete behind the Unit 2 reactor cavity liner.

4.2 <u>Miscellaneous Area</u>

4.2.1 Gap Between Reactor Pressure Vessel Reflective Insulation (RPVRI) and the Biological Shield Wall (See Attachment 2, Miscellaneous Category 2)

- Review the procedures for approval of design changes to non-nuclear safety-related equipment, such as the RPVRI, and make revisions as necessary to assure that such design changes do not adversely affect safety-related systems.

- Review procedures for reporting significant design and construction deficiencies, pursuant to 10 CFR Part 50.55(e), and make changes as necessary to assure that complete evaluations are conducted.

- Provide an analysis which verifies that the cooling flow in the annulus between the RPVRI and the shield wall of Unit 2 is adequate for the as-built condition.

- Verify during Unit 1 hot functional testing that completed modifications to the RPVRI support ring now allow adequate cooling air flow.*

4.2.2 Control of Debris in Critical Spaces Between Components and/or Structures (See Attachment 2, Miscellaneous Category 2; also see Attachment 2, C&S Category 11)

- Identify areas in the plant having critical spacing between components and/or structures that are necessary for proper functioning of safety-related components, systems, or structures in which unwanted debris may collect and be undetected or be difficult to remove;

- Prior to fuel load, inspect the areas and spaces identified and remove debris; and

- Subsequent to fuel load, institute a program to minimize the collection of debris in critical spaces and periodically inspect these spaces and remove any debris which may be present.

4.2.3 Polar Crane Shimming (See Attachment 2, Miscellaneous Category 11)

- Inspect the polar crane rail girder seat connections for the presence of gaps which reduce the bearing surface to less than the width of the bottom flange, and perform an analysis which will determine whether existing gaps are acceptable or require corrective action.

- Determine if additional rail movement is occurring and, if so, provide an evaluation of safety significance and the need for corrective action.

- Perform a general inspection of the polar crane rail and rail support system, correct identified deficiencies of safety significance, and provide an assessment of the adequacy of existing maintenance and surveillance programs.

*This testing has been completed. However, TUEC's analysis of the test results is still underway.

ATTACHMENT 1

LISTING OF TECHNICAL CONCERNS AND ALLEGATIONS

I. Civil and Structural

Allegation Number	Characterization	Category	Page Number
AQC-1	Concrete air entrainment test records were falsified.	8	K-59
AQC-2	Concrete laboratory test records were falsified.	8	K-59
AQC-3	Concrete aggregate tests were falsified.	8	K-59
AQC-4	Equipment required for aggregate testing had not been used.	10	K-71
AQC-5	Improper methods were used to dry coarse aggregate for sieve analysis.	10	K-71
AQC-6	Some of the Unit 1 Containment Building basemat concrete was placed without required testing.	10	K-71
AQC-7	Concrete compressive strength test results were falsified.	8	K-59
AQC-8	Concrete compressive strength test specimens were loaded at an excessive rate.	10	K-71
AQC-9	Recertification examinations for R. W. Hunt inspectors were given open book and examinations were given with answers supplied.	9	K-67
AC-10	Concrete repair following removal of a Hilti bolt was improper.	7	K-57
AQC-11	Acceptable concrete test cylinders were used to represent concrete placements other than those for which the samples were made.	10	K-71
AQC-12	Reinforcing steel (rebar) was installed in the Containment Building without quality control (QC) inspection.	6	K-49

I. Civil and Structural (Continued)

Allegation Number	Characterization	Category	Page Number
AC-13	Diamond core drill bits were loaned for the unauthorized and undocumented cutting of rebar.	15	K-87
AC-14	There was unauthorized cutting of rebar in nonspecific locations.	15	K-87
AC-15	There was unauthorized cutting of rebar during installation of trolley process aisle rails in the Fuel Handling Building.	15	K-87
AQC-16	Rejected aggregate was used in the Unit 1 Reactor Building basemat.	1	K-27
AC-17	(This allegation was reassigned to the Mechanical and Piping Group and is assessed in Mechanical and Piping Category 13, "Metal Shavings from Drilling Fuel Pool Underwater Lamps," under Allegation AP-7.)		
AE-17	Field run conduit, drywall, and lighting installed above the control room were classified nonseismic and were thus inadequately supported.	14	K-83
AC-18	There was unauthorized cutting of rebar in nonspecific locations.	15	K-87
AC-19	Truck drivers added unauthorized quantities of water to concrete used in the basemat.	1	K-27
AC-20	Rejected concrete was placed in the Turbine Building.	1	K-27
AC-21	A batch of concrete with excessive slump was placed in the containment structure.	1	K-27
AC-22	"Bad concrete work" and "sloppy" placement of concrete occurred in unspecified locations.	2	K-31
AC-23	See AC-22.	2	K-31
AC-24	A batch of concrete was placed in the Containment Building dome during a rainstorm without the presence of a QC inspector.	3	K-33

I. Civil and Structural (Continued)

Allegation Number	Characterization	Category	Page Number
AC-25	Voids existed in the concrete wall behind the Unit 1 reactor cavity stainless steel liner.	4	K-39
AC-26	Equipment was set on grout before the grout properly gained its required strength through curing.	5	K-45
AC-27	Rejected and improper material was used in concrete batches. (See AQC-16, AC-19, AC-20, AC-21, AC-47.)	1	K-27
AC-28	Fresh concrete was placed on top of crumbling concrete during construction of a spillway.	4	K-39
AC-29	A pillar, span, or column associated with a spillway was erected 75 to 80 degrees offset.	12	K-79
AC-30	Rebar was omitted from a portion of the Safeguards Building	6	K-49
AC-31	Richmond Insert anchor bolt inserts were installed in Unit 1 at angles not perpendicular to the concrete surface.	5	K-45
AC-32	A 20-ft by 20-ft area of honeycombed concrete in the Unit 1 Auxiliary Building was inadequately repaired.	4	K-39
AC-33	Cracks exist in the Unit 1 concrete basemat and in the floors of other plant buildings.	4	K-39
AC-34	Concrete voids could be detected in building walls by tapping with a hammer and listening for a hollow sound.	4	K-39
AC-35	Concrete was placed in the Safeguards Building basemat and the lowest level floor of the Unit 1 Containment Building during or just before freezing weather.	3	K-33
AC-36	Trash was placed in a form and then covered with concrete.	5	K-45

I. Civil and Structural (Continued)

Allegation Number	Characterization	Category	Page Number
AC-37	Rebar used in the containment structure was not properly inspected upon its receipt at the site. (See AQC-12.)	6	K-49
AC-38	Horizontal tie rebar was missing from the Unit 1 Containment Building wall.	6	K-49
AC-39	Rebar was missing from four column faces along column line EA of the Auxiliary Building at the 807-ft elevation.	6	K-49
AC-40	There was unauthorized cutting of rebar in nonspecific locations.	15	K-87
AC-41	There was poor workmanship in the use of elastic joint filler material, "rotofoam," as a temporary spacer in order to achieve the required airspace between seismic Category I structures.	11	K-75
AC-42	(This allegation is a duplication of allegation AQ-10, "Falsification of Civil QC Records," and has been forwarded to the NRC Office of Investigations (OI) for followup.)		
AC-43	(This allegation reiterated the concerns in AC-26, AC-31, and AC-36.)	5	K-45
AC-44	Cracks existed in the concrete pad beneath the reactor vessel.	13	K-81
AQC-45	Somebody produced incorrect scale readings at the concrete batch plant by leaning on the wires connecting the weight hoppers to the scales.	17	K-95
AQC-46	Midpour test records associated with the Unit 1 Containment Building basemat were falsified.	8	K-59
AC-47	Concrete rejected for being over specification on time to discharge was placed in the Circulating Water Intake Structure.	1	K-27
AQC-48	Concrete test cylinders in the R. W. Hunt laboratory moist room were allowed to dry.	10	K-71

I. Civil and Structural (Continued)

Allegation Number	Characterization	Category	Page Number
AC-49	Rebar was installed upside down in a building near the Unit 2 containment structure.	6	K-49
AC-50	"Soupy" concrete was placed in a slab in the Auxiliary Building during the summer of 1976.	2	K-31
AQC-51	Cadweld tensile test results were recorded during the spring and summer of 1976 without the tests having been performed.	8	K-59
AC-52	Several examples of field-cured cylinders and standard-cured cylinders failed specification requirements. The Schmidt rebound hammer test was then misapplied to resolve these test failures.	3	K-33
AQ-64	Overexcavation and improper fill under the Unit 1 containment structure could invalidate the expected seismic response of the foundation due to changes in properties from the removal of in-situ material.	16	K-93

II. Miscellaneous

Allegation Number	Characterization	Category	Page Number
AM-1	(Issues from this allegation were addressed by the Electrical Group [AE-50 and AE-51]; the Mechanical and Piping Group [AP-24, AP-25, AP-26, AP-27, AP-28, AQW-69, AQW-71]; and the QA/QC Group [AQ-111].)		
AM-2	Nuclear fuel was received onsite before the NRC issued a special materials license.	1	K-97
AM-3	During hot functional testing, expansion caused the reactor pressure vessel reflective insulation to touch the biological shield wall.	2	K-99
AM-4	The Preliminary Safety Analysis Report, Sections 10.2-11 and 10.2-12, contained errors.	3	K-103

II. Miscellaneous (Continued)

Allegation Number	Characterization	Category	Page Number
AM-5	There was a possibility that someone threw radioactive material into the Comanche Peak reservoir.	4	K-105
AM-6	There were cracks in the lower casing of the high pressure turbine.	5	K-107
AM-7	A section of prefabricated pipe was cut "from the pressurizer area."	6	K-109
AM-8	The Unit 1 main condenser tubes were beaten with hammers, were split during belling and flaring, and were improperly rolled.	7	K-111
AM-9	The condenser tube support sheets had holes that were misaligned by 3/8 inch.	7	K-111
AM-10	The turbine-to-condenser tubing was misaligned and then jacked into alignment causing stress.	7	K-111
AM-11	(This allegation was transferred to the Mechanical and Piping Group, Category 43.)		
AM-12	The anchor bolts were damaged during the installation of the component cooling water surge tank.	8	K-115
AM-13	Pumps manufacturered by the Hayward Tyler Pump Company were installed in Comanche Peak safety systems. These pumps may have unidentified deficiencies because of the poor QA program at Hayward Tyler.	9	K-117
AM-14	One of the diesel generators was damaged in May 1982.	10	K-119
AM-15	Shims for the rail suppport system for the polar crane were altered during installation.	11	K-121
AM-16	The polar crane rail moves during crane operation such that large gaps develop.	11	K-121
AM-17	Deficient welds on a missile barrier door were accepted.	12	K-125

II. Miscellaneous (Continued)

Allegation Number	Characterization	Category	Page Number
AM-18	The tube steel used to fabricate supports in the Unit 1 safeguards "796-yard tunnel" was cut at the wrong angle, resulting in excessive gaps for the weld joints between the tube steel and baseplates.	13	K-127
AM-19	The posting requirements for NRC Form 3 were not met from 1977-1982.	14	K-131
AM-20	Material false statements were made by plant management to the Atomic Safety and Licensing Board. (This allegation was transferred to the NRC Office of Investigations for followup.)		
AM-21	There was widespread drug abuse at Comanche Peak, and management did not give proper attention to this problem.	15	K-133
AM-22	TUEC has not analyzed the heating, ventilating, and air conditioning system (HVAC) supports for seismic loads. HVAC components and supports inside containment were not properly considered as missiles. HVAC failure during a postulated accident would allow temperatures to rise to an unacceptable level inside containment.	16	K-137
AM-23(a)	A craft person stated that he had not received instructions about how to rig and handle a large motor-operated valve.	20	K-147
AM-23(b)	The Unit 1 reactor pressure vessel is located 3/16 inch west of the north-south centerline through the containment building.	2	K-99
AM-24	15-foot by 2½-inch stainless steel bars inside the Unit 1 reactor vessel upper internals were damaged and then repaired without proper documentation.	17	K-139

II. Miscellaneous (Continued)

Allegation Number	Characterization	Category	Page Number
AM-25	Internal wires were broken in the polar crane festooned cables, and the polar crane hit some hangers while operating.	18	K-143
AM-26	(This allegation was transferred to the NRC Office of Investigations (OI) for followup)		
AM-27	(This allegation was transferred to OI for followup)		
AM-28	(This allegation is the same as AM-29 below, and was transferred to the Mechanical and Piping Group, Category 39.)		
AM-29	(This allegation was transferred to the Mechanical and Piping Group, Category 39.)		
AM-30	Workers habitually urinated on stainless steel pipe located in the radwaste system.	19	K-145
AM-31	(This allegation was tranferred to the Mechanical and Piping Group, Category 49.)		

ATTACHMENT 2

ASSESSMENT OF INDIVIDUAL TECHNICAL CONCERNS AND ALLEGATIONS IN CIVIL AND STRUCTURAL AND MISCELLANEOUS AREAS

1. Allegation Category: Civil and Structural 1, Inadequate Materials Used in Concrete

2. Allegation Number: AQC-16, AC-19, AC-20, AC-21, AC-27 and AC-47

3. Characterization: It is alleged that the following violations of specifications occurred at various times:

   a. Rejected aggregate was incorporated in the basemat of the Unit 1 reactor (AQC-16).
   b. Truck drivers added unauthorized quantities of water to concrete used in the basemat (AC-19).
   c. Rejected concrete was placed in the turbine generator building (AC-20).
   d. Concrete with excessive slump was placed in containment walls (AC-21).
   e. Some concrete was placed in the Circulating Water Intake Structure after the concrete was rejected for being over specification limit on time to discharge (AC-47).

   AC-27 contained no new allegations; it merely reiterated those already made.

   Allegations AC-19, AC-20, and AC-21 were investigated by Region IV and documented in inspection report 79-09, which was reviewed by the NRC Technical Review Team (TRT) as a step in its own assessment of the allegations.

4. Assessment of Safety Significance: Allegations AC-19, AC-20, and AC-21 appeared in a newspaper article. The identify of the alleger of AC-19 was not disclosed and therefore could not be contacted. Allegations AQC-16 and AC-47 were judged as having sufficient clarity for technical resolution without initial contact between the TRT and the allegers.

   a. The TRT cannot determine whether or not the allegation that "rejected" aggregate was used is valid (AQC-16). The only item in the record was Deficiency and Disposition Report C-446 (December 9, 1976), which stated that an untested pile of aggregate, rather than an unacceptable pile, was used. Therefore, the acceptability of the aggregate is unknown. The alleger also stated that the equipment operator scraped aggregate off the floor of the storage area and dumped it on the conveyer belt so that it bypassed testing. The consequences of this alleged action may be evaluated by the effect on the properties of fresh and hardened concrete. The purpose of controlling aggregate grading is to maintain concrete of uniform workability and strength. A TRT examination of the concrete basemat placement record packages revealed that workability and strength were satisfactory throughout the placement. Less than 3 percent of the concrete was rejected for improper slump, and all concrete tested met the specifications for compressive strength. If any aggregate did not comply with specified grading, the deficiency did not materially affect the concrete properties of the basemat.

b. Construction Procedure CCP 10, para. 4.10.5.6, required the signature of a representative from both the contractor and testing laboratory when water was added to the concrete after it left the plant (AC-19). The batch weights were such that it was possible to add some water to the batch in the ready-mix truck without exceeding the maximum permitted water-cement ratio. The amount of the addition permitted was printed on the batch ticket. However, before the addition was made, the written permission of the testing laboratory was required. The TRT examined all 268 batch tickets and discovered in Concrete Placement Package 101-2781-001, 7-17-75, that 7 batches had water added. For these tickets, the only signature filled in belonged to the contractor representative; none of these tickets was signed by the test laboratory representative. In each case, the volume of water added was within the range permitted. Although the contractor erred in not getting test laboratory approval, the additions should have had no adverse effect on the concrete. This error, however, indicated a breakdown in the quality control system. A TRT examination of test results indicated that all were within specification guidelines: slump values ranged from 1 inch to 2-3/4 inches; air content was from 2.0 percent to 3.2 percent; and 28-day compressive strengths ranged from 5340 psi to 6671 psi. In addition, the TRT examined parts of the basemat which were still accessible. While only a small portion could be examined visually and this portion did not necessarily include any batches with added water, the portion examined was in excellent condition.

c. The alleger did not indicate where in the turbine building the alleged infraction occurred (AC-20). The building contains over 700 small concrete placements, and all were available for examination. The TRT examined a random selection of 65 concrete placement packages and found no irregularities. However, the turbine generator building is a nonsafety-related structure. The Final Safety Analysis Report, Sec. 3.2, "Classification of Structures, Components, and Systems," indicates that the turbine-generator building is not a seismic Category I structure. Its structural failure would not affect safety during a safe shutdown earthquake; therefore, the activity alleged to have occurred would not affect the safety of the plant.

d. The alleger claimed that a batch with a slump of 4-1/4 inches was placed (AC-21). The slump requirement in Gibbs & Hill (G&H) Specification 2323-SS-9, Revision 4, Section 5.2, states:

> A tolerance of up to 1 inch above the indicated maximum shall be allowed for individual batches provided the average of all batches tested or the most recent 10 batches tested, whichever is fewer, does not exceed the maximum limit, i.e., 4 inches. Whenever the measured slump exceeds the indicated maximum by more than 1/4 inch, successive batches or truck loads as deposited shall be measured until the slump is within the maximum limit.

Thus, placement of a batch with a 4¼-inch slump was permitted as long as the average of all batches or the most recent 10 batches did not exceed 4 inches and individual batches did not exceed 5 inches. A single high slump batch, provided the slump does not exceed 5 inches, cannot constitute a violation of specifications.

e. The TRT reviewed 51 (37%) concrete packages out of the 140 concrete packages for the Circulating Water Intake Structure which is a nonsafety-related structure (FSAR Vol. IV, Sec. 3.2) (AC-47). Of the 51 reviewed, 13 batches of concrete were rejected, 9 for test failure (air, slump, temperature), and 4 for being over the specification limit on time to discharge. It was noted on the batch ticket of each rejected batch of concrete where the concrete was dumped. None of the rejected batches was placed in the circulating water intake; they were placed in temporary slabs which the contractor was placing at the time.

5. Conclusion and Staff Positions: The allegations were found to have no structural safety significance.

Based on a review of pertinent documentation, test results, and the concrete placement packages, the TRT concludes that if nonconforming aggregate was used in the basemat of the Unit 1 reactor, it did not adversely affect its concrete properties. The only indication of water addition found by the TRT was within the stipulated limits, thus ensuring that there was no adverse effect on the concrete. However, the absence of laboratory signatures on batch tickets represents a failure to follow QA/QC program requirements. The results of the evaluation pertaining to the lack of laboratory signatures will be further assessed as part of the overall programmatic review concerning procedures addressed under QA/QC Category 6 "QC Inspection." Therefore, the final acceptability of this evaluation will be predicated on the satisfactory results of the programmatic review of this subject. In its examination of 65 random samples of concrete placements in the turbine building, which is a nonsafety-related structure, the TRT found no evidence of irregularities. Batch placements were within tolerances specified by G&H, and the TRT found no documentation that these slump requirements had been violated. The placement of a single batch of concrete with a 4¼-inch slump does not constitute a violation of specifications. The batch tickets state that none of the rejected concrete batches was placed in the circulating water intake; therefore, the TRT concludes that the allegation is without foundation.

The alleger of AC-19 was not identified so that the TRT could not conduct a closing interview. The TRT is attempting to contact the individual who made allegation AC-21. The individual who made allegation AQC-16 did not wish to meet any further with the TRT and will be informed of the pertinent TRT findings by letter. The individual who made allegation AC-20 declined to be interviewed by the TRT and will also be informed of the pertinent TRT findings by letter. The alleger of AC-47 could not be located for a closing interview.

6. Actions Required: None.

1. Allegation Category: Civil and Structural 2, Concrete Placements

2. Allegation Number: AC-22, AC-23 and AC-50

3. Characterization: It is alleged that "bad concrete work" and "sloppy" placement of concrete occurred at the Comanche Peak Steam Electric Station (CPSES) (AC-22, AC-23). It is also alleged that "soupy" concrete was placed in a slab in the Auxiliary Building in the summer of 1976 (AC-50).

4. Assessment of Safety Significance: The individual making allegations AC-22 and AC-23 was interviewed by the NRC Technical Review Team (TRT). Allegation AC-50 was judged as having sufficient clarity for technical resolution without initial contact between the TRT and the alleger.

   In testimony at an Atomic Safety and Licensing Board (ASLB) hearing, the first alleger did not identify a particular structure or concrete placement that had "bad concrete work" or that exhibited "sloppy" placement of concrete. To adequately encompass the concerns raised in this allegation, the TRT reviewed random samples of concrete placement packages from three safety-related buildings to determine if the allegations were valid.

   A review of 14 packages from the Auxiliary Building and 3 packages each from the Unit 1 Safeguards Building, the Unit 1 Containment Building (exterior), and the Unit 2 Containment Building (exterior) revealed that the quality control (QC) inspector accepted the forms and reinforcing steel placement prior to each concrete placement. Of the 23 placement packages reviewed, 10 had nonconformance reports (NCRs) related to concrete placement. One package had four NCRs, two other packages had two NCRs each, and the remaining seven each had one NCR. Seven NCRs were resolved with the designation "use-as-is," seven with "repair," and one with "reject." The seven placements indicating "repair" were for concrete honeycombing; the one indicating "reject" was for the removal of concrete from a small pad. In addition to a records review, the TRT performed a walkthrough inspection of the safety-related buildings. The defects treated in these NCRs are visible from the surface and were examined by the TRT. The TRT concluded that there was no degradation in quality in any of the observable concrete surfaces.

   The TRT also interviewed two QC inspectors at Comanche Peak who were concrete placement inspectors on some of the concrete placements in the Auxiliary Building reviewed by the TRT. Both QC Inspectors stated that they were not cognizant of any "bad concrete work" and/or "sloppy" placement of concrete at CPSES. They stated that all personnel with construction and concrete placement responsibilities would meet prior to each placement to resolve any potential problems. They stated that for the concrete placements they were involved with, the work was done in accordance with project procedures and other pertinent requirements. They also stated that placement crews cooperated with requests from QC personnel. The individual who made the allegations discussed above was contacted by the TRT to inform him of the TRT's finding. The alleger expressed his satisfaction with respect to the TRT's disposition of his allegations.

To investigate the allegation of "soupy" concrete in an Auxiliary Building slab, the TRT examined the following three placement packages, which included all the slab concrete placed during the summer of 1976: 002-7785-001, 002-2790-003, and 002-2790-004. The TRT noted that before placing the first section, the contractor requested permission, which was granted, to place mortar rather than concrete in one area heavily congested with reinforcing bars. This might have been the "soupy" concrete cited by the alleger. During placement of the three sections, five batches of concrete were rejected for excessive slump. In four of these cases, two or three cubic yards had been placed per the requirements of the ASTM Standard Method for sampling fresh concrete (ASTM C 172). ASTM C 172 requires that samples be taken at two or more regularly spaced intervals during discharge of the middle portion of the batch; and that samples not be taken from the very first or last portions of the batch. However, 70 to 80 percent of each batch was discarded. The concrete already in place was left in the forms. This type of occurrence is considered a normal procedure in concrete placement work and is judged to have no effect on safety.

5. Conclusion and Staff Positions: The TRT evaluated the allegations by reviewing a random sample of concrete placement record packages, by interviewing two former concrete placement inspectors, and by conducting a walkdown inspection of finished concrete work in three safety-related structures. This level of evaluation was deemed necessary to adequately encompass the potential scope of the allegations, which were not specific about where at Comanche Peak the "bad" and "sloppy" concrete work had been performed. In its records review, the TRT found some discrepancies in concrete placements that were identified and resolved by established QC procedures. However, the discrepancies found are not uncommon in concrete work; the TRT walkdown provided evidence that the discrepancies were resolved in that the concrete shows no degradation. The TRT also investigated the specific allegation concerning "soupy concrete" by reviewing all the relevant concrete placement packages and found the allegation to be without safety significance. The TRT found that mortar had been authorized in lieu of concrete for a small portion of the structure. Accordingly, these allegations have neither safety significance nor generic implications.

   The individual making Allegations AC-22 and AC-23 has indicated his satisfaction with the TRT disposition of his allegations. The alleger of AC-50 has not been located. The TRT is still trying to locate him for a closing interview.

6. Actions Required: None.

1. Allegation Category: Civil and Structural 3, Poor Weather Conditions

2. Allegation Number: AC-24, AC-35 and AC-52

3. Characterization: It is alleged that the placement of some concrete took place under the following adverse weather conditions: (a) during a rainstorm and without the approval of quality control (QC) personnel (AC-24) and (b) during or immediately before freezing weather (AC-35). It is further alleged that (c) there are several examples of field-cured cylinders which failed specification requirements, that some standard-cured cylinders failed specification requirements, and that the Schmidt rebound hammer test was misapplied in resolving problems created by these deficiencies (AC-52).

4. Assessment of Safety Significance: Allegation AC-24 was the subject of testimony given by Region IV inspectors, but the identity of the alleger, was not revealed. The TRT attempted to determine the alleger's identity but could find no record of it. Allegation AC-35 was judged as having sufficient clarity for technical resolution without initial contact betweer the NRC Technical Review Team (TRT) and the alleger. The TRT interviewed the alleger of AC-52.

   a. In assessing the allegation concerning placement during a rainstorm (AC-24), the TRT examined concrete placement package 101-8805-013 for a placement on the dome of the Unit 1 Containment Building. This package indicated that the final batch of concrete placed on the evening of January 18, 1979, was batched at 5:59 p.m.; that only about 300 of the required 450 cubic yards had been placed; that the crew abandoned the placement in a heavy rain at 7:30 or 8:00 p.m., leaving a gap with a 30-foot radius in the middle of the placement; that concrete batching started again at 8:00 a.m. on January 19; and that the lift was topped out at 12:21 p.m. There is no account of any irregularity during the shutdown. However, the craft personnel general foreman for the placement reported that the attempt to cover the partially completed concrete with plastic to protect it from the rain was not completely successful; that about a half cubic yard of concrete was washed out before the crew got the situation under control; that at about 10 p.m. he went to the batch plant, which was now empty because of the hour, dry-batched a half-cubic yard to the correct proportions, mixed it in two batches in the concrete laboratory mixer, and placed it on the dome. At this time, all quality control personnel had gone home and were not available to approve or oversee the operation. This sequence of events was not refuted by the NRC Region IV investigation (inspection report 79-11) of this incident and is apparently correct. The TRT interviewed the author of the Region IV inspection report.

   The action constitutes a violation of 10 CFR 50, Appendix B, Criterion X and indicates a partial breakdown of the quality control system. Following the completion of the dome, and after learning of the allegation of a violation, Brown & Root engaged Muenow and Associates to make an ultrasonic investigation of the portions of the dome potentially affected by the rainstorm. They also engaged Erlin, Hime, and Associates to interpret the Muenow report. The incident and the investigation are discussed extensively in nonconformance report (NCR) C-1418. The TRT reviewed NCR C-1418, "Final Report on the Concrete Evaluation in Dome Roof Section of Comanche Peak Unit 1," by Richard Muenow of Muenow and Associates, and "Discussion of Final Muenow Associates Report, Comanche Peak Steam Electric Station, Reactor #1 Dome Concrete Testing for Texas Utilities Service, Inc.," by B. Erlin of Erlin, Hime, and Associates. The investigation revealed some minor discontinuities at 4 to 5 inches from the surface, and at 10 to 12 inches from the surface, with a few voids at a maximum of less than ½ inch. Correlation of pulse velocity data on the dome with test cylinders containing the same materials indicated compressive strength in excess of 4,000 psi; this high pulse velocity, combined with the relative absence of voids, indicated a density in excess of 140 pounds per cubic foot. Accordingly, the structure as built satisfies the design requirements in the Final Safety Analysis Report. More convincing evidence of the acceptability of the dome concrete was provided by TUGCO's "Final Report on Structural Integrity Test for Unit 1 Concrete Containment Structure," CPDA-31, 792. The containment structure met all criteria for displacement and cracking control as well as structural rebound when subjected to 115 percent of design pressure.

b. The allegation concerning concrete placed in freezing weather (AC-35) was in connection with the Safeguard Building basemat and the lowest level floor of the Unit 1 containment structure. The TRT reviewed in detail the relevant concrete placement packages, namely 105-2773-001 and 101-2808-001.

   (1) Placement package 105-2773-001 reports on the Safeguard Building basemat, which was placed on December 31, 1975. All surface temperatures in the records comply with the specifications. However, it is alleged that on the seventh day of curing, when the ambient temperature dropped to 18°F, a portion of the concrete in place was not protected by insulation. Brown & Root interoffice memo IM 4152 stated that all concrete was well covered with insulation except the edges, where it was difficult to place insulation because of protruding dowels, but that a careful examination of the concrete showed no evidence of damage caused by freezing. Of the 15 field-cured specimens tested at 28 days, 12 failed the criterion of equalling or exceeding 0.85 of the laboratory-cured specimens. Of these, two failed the alternate criterion of exceeding the design strength by 500 psi. However, all results exceeded the design strength of 4,000 psi. The fact that 2 out of 12 failed to meet specification requirements is not serious for concrete such as this, which was not loaded at an early age. The results of field-cured cylinder tests indicated that the cold weather slowed the strength gain, but that the protection was adequate to attain the design strength in 28 days. Subsequent warmer temperatures provided all the strength required by the specifications. To compare the concrete near the dowels with concrete whose protection was not in doubt, the R.W. Hunt Co. ran Schmidt hammer tests on both the

suspect concrete and the acceptable concrete at an age of 4 months. The results are recorded in HCP reports 10664 and 10849, which were inspected by the TRT. For both series of tests rebound numbers ranged from 39 to 46. The concrete on the edge adjacent to the dowels, which was difficult to protect, is acceptable for the following reasons: (1) it was not exposed to freezing temperatures for 6 days following its placement; (2) concrete at that age should not be damaged by freezing; and, (3) Schmidt hammer readings were the same on suspect concrete as on well-protected concrete.

(2) Placement package 101-2808-001 reports on the concrete in the Unit 1 containment structure, which was placed on December 30, 1976. On the evening following the placement, the ambient temperature dropped below 20°F. The records showed a concrete surface temperature as low as 42°F during the first day and no surface temperatures below 50°F on subsequent days, in spite of the fact that ambient temperatures as low as 12°F were measured. The protection, as indicated by the records, complied with specifications. However, the allegation was triggered by an event detailed in Brown & Root (B&R) interoffice memo IM 7700. During the first evening, a TUEC QC inspector measured a surface temperature of 21°F. The B&R QC inspector noted that the TUEC inspector used an uncalibrated thermometer with a large range and took the reading in such a manner that the thermometer was not protected from the air so that in the B&R QC inspector's opinion, the TUEC inspector was measuring ambient temperature instead of the concrete surface temperature. Although the two discussed the adequacy of the technique, and a picture was taken of the technique, the records did not indicate that the matter was ever resolved.

To evaluate the condition of concrete alleged to have been exposed to freezing temperatures, the R.W. Hunt Co. ran Schmidt hammer tests on the suspect concrete and on concrete whose integrity was not in doubt. The results are in HCP report 22014, which was examined by the TRT. Rebound numbers for suspect areas ranged from 25 to 35, and in sound areas from 27 to 36. The differences are not significant.

c. The allegation concerning field-cured test cylinders, standard-cured test cylinders, and Schmidt rebound hammer tests (AC-52), is contained in the attachments to a letter, dated September 20, 1984, to Thomas Ippolito, NRC, from Mrs. Juanita Ellis, President of the Citizens Association for Sound Energy (CASE). The allegation states, "Based on a review of documents attached and already in the record, it is apparent that the quality and compressive strength of the concrete at Comanche Peak is indeterminate at best, and, in some cases appears to be deficient." This observation is presumably supported by Attachment D to the letter, which lists 36 test cylinders in 18 placements with laboratory-cured strengths below 4000 psi. The alleger was interviewed, and he stated that he was under the impression that the concrete was designed for a strength of 4000 psi. The TRT reviewed the records and found that all the cited concrete was of designations C-301, C-302, C-305, or C-306, all of which have a design strength of 2500 psi. The lowest reported strength was 3267 psi. Thus, all these strengths met the strength specification by a wide margin. Furthermore, the TRT has discovered no standard-cured test cylinders in safety-related structures which failed the strength requirement.

The allegation that some field-cured test cylinders failed to meet specification requirements is correct. The project specifications, 2323-SS-9, paragraph 7.3, cited the requirements of ACI-318, the American Concrete Institute Building Code. These requirements are very conservative and are intended for building construction where slender flexural members are required to sustain a large portion of their design load at an early age. The requirement is that cold-weather protection shall be improved when the 28-day strength of field-cured cylinders is less than 85% of the strength of laboratory-cured cylinders. This requirement is more restrictive than is necessary for a massive structure. The definitive American Concrete Institute guidance on cold-weather protection is provided in ACI 306R-78, "Cold Weather Concreting." That document states that items such as foundations, substructures, and massive sections not subject to early load, which will be subjected to favorable curing temperatures prior to receiving the full design load, should be protected for 2 days if they are not subject to freezing in service and 3 days if they are subject to freezing. Protection is defined as maintenance of a temperature of 55°F for sections thinner than 12 inches, 50°F for sections 12 to 36 inches thick, 45°F for sections 36 to 72 inches thick, and 40°F for sections thicker than 72 inches. No strength requirements are stipulated.

All the field-curing deficiencies cited in Attachment D to the CASE letter, with three exceptions, fall into this less stringent category. The three exceptions are a slab in the Auxiliary Building, in which the field-cured strength was 3891 psi, and two cylinders representing slabs in the Safeguards Building with strengths of 3407 and 3956 psi. In these cases, the design strength was 4000 psi. The first and third had strengths sufficiently close to 4000 psi to eliminate any concern for safety. The second was in a region tested by the Schmidt rebound hammer and found to be equal in quality to sections of concrete whose quality was not in doubt. Even though more lenient criteria could reasonably have been established for much of the concrete, cold-weather protection was generally quite good. The ACI criteria for massive structures can produce field-cured strengths as low as 50% of the design strength at 28 days if the concrete is maintained at 35°F after protection is terminated. In contrast, most test cylinders at Comanche Peak exceed the design strength. Of 108 cylinders with a design strength of 4000 psi cited in Attachment D as failing to comply, 94% exceeded 3000 psi and the lowest strength was 2477 psi. Of 17 field-cured test cylinders failing to meet the design strength of 2500 psi, 14 exceeded 2000 psi, and the lowest strength was 1820 psi. It also may be noted that field-cured cylinders usually underestimate the strength of the in-place concrete they represent because they are not as massive and, therefore, benefit less from heat produced by hydration of cement during the curing process.

The allegation questioned the use of the Schmidt rebound hammer for qualifying sections of concrete in which field-cured test cylinders failed to meet specifications. The above discussion makes the issue moot except for the single cylinder in a slab of the Safeguards Building. The use of the Schmidt rebound hammer has general acceptability and is specifically permitted by the Comanche Peak construction specifications as an aid in evaluating concrete strength in place, as discussed below. ASTM C-805 states, "The rebound number determined by this method may be used to assess the uniformity of concrete in situ, to delineate zones or regions of poor quality...." Paragraph 7.3.e of the project specifications states, "Evaluation of test results shall be in accordance with Section 17.1, 17.2, and 17.3 of ACI 301." Section 17.3.1 of ACI 301 states, "Testing by impact (Schmidt) hammer, soniscope, or other nondestructive device may be permitted by the architect/engineer to determine relative strengths at various locations in the structure as an aid in evaluating concrete strength in place or for selecting areas to be cored. Such tests, unless properly calibrated and correlated with other test data, shall not be used as a basis for acceptance or rejection." Hammer results are normally not permitted as a substitute for laboratory-cured test cylinders, which form the basis for acceptance of the concrete. They may be used to judge the adequacy of protection or to determine when a portion of a structure may be safely loaded. The ACI Building Code and the Comanche Peak specifications do not provide for the rejection of concrete on the basis of low-strength, field-cured cylinders. They merely require that protection be improved and that critical elements be cured for a longer period of time before being loaded. With the exception noted above, the low field-cured strengths were not in critical elements. The statement in Attachment D that all retesting which had been promised had not been carried out is a quality assurance matter, not a safety problem, and it will be investigated by the TRT QA/QC group. The statement that Schmidt hammer tests were not conducted on sections of concrete when both field-cured and laboratory-cured cylinders were below 4000 psi does not appear to be pertinent since there were no sections cited where both field and laboratory results were below the design strength.

5. Conclusion and Staff Positions: Although these allegations are true, they do not have structural safety significance.

   (a) The Unit 1 dome was proved sound both by ultrasonic testing and by structural integrity testing.

   (b) Sections of concrete alleged to have been exposed to freezing temperature at an early age were shown by in-place strength tests to have substantially the same strength as concrete whose protection was not in doubt.

   (c) The field-cured test cylinders demonstrated adequate protection for the type of concrete placed, with the exception of one slab in the Safeguards Building, which was shown by Schmidt hammer testing to be adequate.

Accordingly, these allegations have no structural safety significance. However, the results of the evaluation pertaining to the placement of concrete without QA/QC involvement, the response for improving protection when field-cured cylinders showed inadequate strength, and the failure to carry out promised retests will be further assessed as part of the overall programmatic review concerning procedures addressed under QA/QC Category 6, "QC Inspection." Therefore, the final acceptability of this evaluation will be predicated on the satisfactory results of the programmatic review of this subject. Any adjustments to the existing conclusion of this evaluation resulting from the programmatic review will be reported in a supplement to this SSER.

The TRT was unable to establish the identity of the individual who made allegation AC-24. The TRT cannot locate the alleger of AC-35 and has closed the allegation. The TRT has previously interviewed the alleger of AC-52, and a closure interview with the alleger is scheduled.

6. Actions Required: None.

1. Allegation Category: Civil and Structural 4, Concrete Voids and Cracking

2. Allegation Number: AC-25, AC-28, AC-32, AC-33 and AC-34

3. Characterization: It is alleged that the following concrete deficiencies occurred at Comanche Peak Steam Electric Station (CPSES):

   a. Hollow places existed in concrete behind the stainless steel liner of the Unit 1 reactor cavity (AC-25).

   b. Fresh concrete was placed on top of crumbling concrete during the construction of the spillway (AC-28).

   c. The repair of a 20-foot x 20-foot honeycombed area located in the Unit 1 Auxiliary Building was inadequate (AC-32).

   d. Cracks existed in the concrete reactor cavity wall of Unit 1 and in floor slabs in the plant buildings (AC-33).

   e. There are numerous concrete voids in building walls that can be located by tapping the walls with a hammer and listening for a hollow sound (AC-34).

   Allegation AC-25 was investigated by Region IV and documented in inspection reports 80-08 and 80-11, which were reviewed by the TRT as a step in its own assessment of the allegation.

   In addition to these allegations, the Region IV resident inspector requested that the TRT review the following possible reportable design deficiencies involving concrete placing problems.

   f. Reportable Design Deficiency Concerns:

      (1) A void was identified in the Unit 1 Reactor Building Steam Generator Compartment Wall.

      (2) On concrete placement 002-7810-002 at the 810-foot elevation of the Unit 2 Auxiliary Building, embedded foreign material was located with a flex drill.

4. Assessment of Safety Significance: The NRC Technical Review Team (TRT) interviewed the individuals who made allegations AC-25 and AC-28. Allegations AC-32, AC-33, and AC-34 were made by former Brown & Root employees. The TRT attempted to determine their identity but was unable to do so.

   a. The alleger originally stipulated that the hollow places were located behind the stainless steel liner of Unit 1 reactor cavity, but when interviewed by the TRT, he stated that he meant Unit 2. In assessing the allegation, the TRT interviewed the TUEC chief structural engineer who stated that when forms were removed from one section of the Unit 2 reactor cavity structure, honeycombed areas were discovered on the side of the structure accessible to visual examination. Because of the concern that the honeycombing indicated inadequate concrete consolidation in this section and because the possibility existed that there might also be voids on the opposite side of the reactor cavity wall which were not accessible to visual examination, TUEC examined that section of the concrete wall ultrasonically. The examination revealed the existence of voids behind the stainless steel liner. Their existence and the required repair procedures are documented in Design Change Authorization (DCA) No. 6663. Repairs were being performed by TUEC at the time of the TRT review.

b. There are two spillways at the CPSES, one located near the safe shutdown impoundment (SSI), and the other located at the Squaw Creek Dam. The allegation did not specify which was intended, but the SSI spillway was eliminated from consideration because it was constructed after June 1978, while the period cited in the allegation was 1976 and 1977. The Squaw Creek Dam spillway was constructed from August 1976 to January 1977.

   The TRT review of placement documentation indicated that the Squaw Creek spillway was placed in a single "lift"; therefore, no new concrete could have been placed on hardened or crumbled concrete.

   During the interview with the alleger, it became apparent to the TRT from the types of placements being described that he had a general concern about the adequacy of cold weather placement practices during construction of the Squaw Creek Dam and appurtenant structures. However, he was unable to identify a specific spot where specifications were violated. The TRT examined documentation for cold weather protection for several placements during its investigation of other CPSES allegations. Those examinations confirmed that cold weather protection was adequate. Furthermore, the Final Safety Analysis Report, Section 3.2, "Classification of Structures, Components, and Systems," indicates that the Squaw Creek Dam is not a seismic Category I structure. Its failure would not affect safety during a safe shutdown earthquake.

c. The concrete honeycombing referred to in the allegation is documented in nonconformance report (NCR) C-1034. The architect-engineer's direction was to remove the honeycombed area down to sound concrete and then fill the void area with dry-pack concrete or small size coarse aggregate concrete, all in accordance with a standard, engineer-approved, repair procedure for such work. The TRT reviewed the repair procedure used (QI-QP-11.0-5) and believes it is adequate to properly repair the affected area. The repair is documented in Region IV Inspection Report 50-445/79-26. The NRC Resident Reactor Inspector (RRI) observed various phases of the repair work from August 1978 through January 1979, when the repair was finally completed. The RRI noted that the work was being done in an acceptable manner and in accordance with the approved instructions.

   The TRT inspected documentation pertaining to the honeycombed area in the Auxiliary Building for concrete placement 002-7852-007 and verified that the area had been repaired. The TRT review of this concrete placement package revealed no documentation discrepancies concerning the repair.

d. The existing cracks in the Unit 1 concrete reactor cavity wall have been the subject of a great deal of attention by the NRC and the designer. They have been documented in numerous NCRs, such as NCR C-650 and NCR C-1034. The TRT reviewed a random sample of the concrete placement packages for the Unit 1 Containment Building, Auxiliary Building, and both Safeguards Buildings, and found no evidence of specification violations during the concrete placement. The TRT also inspected the cracks documented in NCRs 1034 and 650. The crack documented by NCR C-1034 is a small hairline crack, caused by shrinkage or thermal effects, that is so small that it cannot impair structural behavior and capacity. Cracks documented by NCR C-650 are evaluated in Civil and Structural Category 13.

e. The NRC Resident Reactor Inspector (RRI) at Comanche Peak Steam Electric Station (CPSES) investigated this allegation (Region IV Inspection Report 50-445/80-16 and 50-446/80-16). The RRI learned that the alleger had worked at the site for 5 weeks in early 1980 in the Unit 1 Safeguards Building at the 790-foot elevation. The RRI found two locations at that elevation where a hollow sound could be obtained by tapping a wall with a hammer. He informed TUEC of this condition, and they found several more locations in the same general vicinity, all at the 790-foot elevation. Each area was marked and excavated to approximately 2 inches, that is, to the depth behind the first layer of reinforcing steel. The RRI observed several excavations and saw nothing abnormal about the concrete. He also queried the craft personnel who were excavating when he was not present and was informed that all excavations revealed nothing except uniformly solid concrete. The RRI tapped the concrete after it had been excavated to a depth of approximately 4 inches and could no longer detect a hollow sound. The allegation apparently was based on the premise that what the alleger interpreted as a hollow sound indicated a void in the wall. Excavations of the areas in question revealed no voids in the concrete.

f. (1) This item was not the subject of an allegation. The TRT reviewed its disposition because it involved an issue similar to those raised in other CPSES allegations.

Nonconformance report (NCR) C-82-00858, which was reviewed by the TRT, indicates that a void did exist in the generator compartment wall of the Unit 1 Reactor Building. As part of the NCR resolution, the matter was reported to Gibbs & Hill (Office Memorandum CPPA-21495, July 20, 1982) and they concluded that the wall would perform both its structural and radiation shielding functions whether or not the void was filled. However, to ensure that no safety issue could be raised, Brown & Root filled the void with nonshrink grout in August 1982, as documented in Inspection Report IR-C-6682. The TRT agrees that in its repaired state the wall presents no safety problem.

(2) This item was not the subject of an allegation. The TRT reviewed its disposition because it involved an issue similar to those raised in other CPSES allegations.

The deficiency was documented in NCR C-82-01432, which was reviewed by the TRT. The TRT learned that a worker drilling holes for anchor bolts in a floor of the Auxiliary Building encountered an apparent void and debris. The debris appeared to be plywood chips. A Brown & Root examination of the area revealed that the drill had hit an embedded drain pipe and had removed some of the foam insulation wrapped around the pipe per drawing MI-781. The driller had apparently mistakenly identified the foam as plywood. The disturbed insulation and concrete were then replaced, as documented in Brown & Root Inspection Report IR-C-7035. The TRT reviewed the Inspection Report and determined that the area was repaired in an acceptable manner.

5. <u>Conclusion and Staff Positions</u>:

a. The allegation of hollow places in concrete behind the stainless steel liner of the Unit 2 Reactor Cavity is true and cannot be closed at this time. The area is currently undergoing repairs; the repairs must be inspected and approved by the NRC Resident Inspector before the TRT can determine that this issue has been adequately resolved.

The following allegations and concerns were found to have no structural safety significance.

b. The TRT reviewed documentation for several placements done in cold weather and concludes that the protection was adequate. In addition, the allegation has no safety significance, since the spillway in question is not safety related.

c. The allegation of honeycombing in the Unit 1 Auxiliary Building is true and the repairs made were in accordance with approved procedures. Therefore, the allegation has no structural safety significance.

There are numerous NCRs dealing with honeycombed concrete. Their evaluation and subsequent concrete repairs are well-documented and did not result in allegations of improper construction except for those discussed herein. The quality assurance system apparently was adequate in documenting these repairs. However, there appears to have been a breakdown of quality control overseeing the consolidation of concrete as evidenced by the numerous NCRs and allegations AC-25 and AC-32. The results of the evaluation pertaining to inadequate consolidation of concrete will be further assessed as part of the overall programmatic review concerning procedures addressed under QA/QC Category 6 "QC Inspection." Therefore, the final acceptability of this evaluation will be predicated on the satisfactory results of the programmatic review of this subject. Any adjustments to the existing conclusion of this evaluation resulting from the programmatic review will be reported in a supplement to this SSER.

d. While the allegation of cracking in the concrete basemat is accurate, it is not correct to assume that detrimental structural consequences will result from the cracks. The structures are designed to tolerate cracks of the magnitude and location of those found.

e. The allegation of numerous concrete voids was not substantiated.

f. (1) The reported void in the generator compartment wall of the Unit 1 Reactor Building is true. The void was filled even though it did not require filling from the standpoint of adequacy of design. The TRT determined that the wall in its repaired condition is safe.

   (2) The area reported as containing unusual material in the concrete was adequately repaired so that this condition will have no impact on safety.

The TRT will inform the individual who made allegation AC-25 of the TRT's findings by letter. The alleger of AC-28 has been notified by letter of the TRT disposition of his allegation. The allegers of AC-32, AC-33 and AC-34 are former B&R employees. The TRT was unable to establish their identity.

6. <u>Actions Required</u>: The repairs and the repair documentation to the honeycombing discussed in Item a must be inspected/reviewed and approved by the NRC Resident Inspector before the TRT can determine whether this issue has been adequately resolved. The successful completion of the repairs shall be reported to the TRT and will be verified by the NRC Resident Inspector prior to low-power operations.

1. Allegation Category: Civil and Structural 5, Miscellaneous Concrete

2. Allegation Number: AC-26, AC-31, AC-36 and AC-43

3. Characterization: It is alleged that the following irregularities occurred in connection with concrete construction:

   a. Equipment was set on grout before the grout properly gained strength through aging (AC-26).
   b. Hanger inserts were installed at improper angles (AC-31).
   c. Trash in the bottom of a form was covered with concrete (AC-36).

   AC-43 did not include any new allegations; it merely reiterated those made in AC-26, 31, and 36.

4. Assessment of Safety Significance: The NRC Technical Review Team (TRT) interviewed the alleger of AC-31. Allegation AC-26 was judged as having sufficient clarity for technical resolution without initial contact between the TRT and the alleger. Allegation AC-36 was the subject of testimony given by Region IV inspectors, but the identity of the alleger was not revealed. The TRT attempted to determine the alleger's identity but could find no record of it.

   a. Allegation AC-26 concerns grouting of steel plates which were baseplates for the frames used to support parts of the internal assembly in Unit No. 2 when they were removed from the reactor pressure vessel. If the grout were damaged by the steel plate being loaded prematurely, the damage would occur immediately, while the grout was weak. If the grout survived the loading operation without damage, it probably would not suffer damage in use, since it gains strength rapidly while it is fresh and at a decreasing rate as it ages.

      All elements of the internal assembly were located at the 860-foot. elevation. The TRT inspected all the grouted plates at the 860- and 862-foot elevations and found no evidence of grout failure. While the allegation may be true, all the grout survived the initial loading without damage. If this allegation is true, a quality control issue exists. The TRT Civil and Structural Group did not look into the QA/QC aspects of this allegation.

   b. It is alleged in AC-31 that Richmond anchor bolt inserts were installed between the 860- and 905-foot elevations in Unit 1 at angles not perpendicular to the concrete surface and that this condition was compensated for by use of tapered washers. The allegation referred to discrepancies as great as ten degrees. The allegation was addressed in NRC Inspection Report 50-445/83-27, which was reviewed by the TRT.

      The TRT found that Brown & Root Procedure CP-CPM 9.10, "Fabrication of ASME-Related Component Supports," stated in Section 3.3.2 that:

> Surfaces of bolted parts in contact with the bolt or nut shall have a slope of no more than 1:20 with respect to a plane normal to the bolt axis. Where the surface of a high strength bolted part has a slope of more than 1:20, a beveled washer shall be used to compensate for the lack of parallelism.

Thus, inserts may depart 3 degrees from perpendicularity without any compensation and may depart further than 3 degrees if beveled washers are used. The procedure mentioned no upper limit on lack of perpendicularity. It did, however, stipulate that the predrilled holes in the tubular steel hanger safety-related supports may not be enlarged without prior approval.

The TRT inspected 150 anchors between the 860- and 905-foot elevations. The inspection consisted of a visual check of perpendicularity of the "as-installed" anchors, the occurrence of non-perpendicular inserts without the compensation of using beveled washers, the maximum extent of insert deviation from perpendicularity, and the evidence of hole enlargements. Two were found to deviate from perpendicularity by more than 1:20; in these cases beveled washers were used. No hole enlargements were found. Thus, the TRT found no violation of the installation procedure. The allegation correctly asserts that some anchor inserts were not perpendicular to the concrete surface; however, that in itself did not constitute a violation of procedure.

c. Allegation AC-36 is concerned with trash from a Christmas party in December 1978, that was thrown into the form and was covered with concrete that was being placed on one of the two containment structures. The alleged incident is extensively discussed in NRC Inspection Report (IR) 50-445/79-20, which was reviewed by the TRT. Interviews with alleged participants, which were reported in IR 50-445/79-20, cast considerable doubt as to whether the party actually occurred. It was established in the inspection report that during December 1978 the alleger was at the project only on December 2, 3, and 4.

The TRT obtained a printout of all concrete placements on the containment structures, and determined that the only placement which occurred during the period in question was on the dome of Unit 1 on December 3, 1978. The TRT examined concrete placement package 101-8805-002, which contained a complete narrative of the placement operation by the placement inspector. Nothing unusual was noted, and both the formwork and cleanliness were checked as "satisfactory" on the checkout card. If anything unusual, such as dumping of trash, did take place, the structural integrity of the dome concrete was not compromised. The dome was proven to be adequate both in strength and in structural capacity, as indicated by the Unit 1 structural integrity test discussed in Civil and Structural Category 3.

The TRT interviewed the individual who had raised the concern regarding the installation of anchor bolt inserts. This individual did not agree with certain TRT findings and provided the TRT with more information regarding his concern. The TRT investigated his concern further and scheduled another interview, but he declined to appear.

5. Conclusion and Staff Positions: The TRT concludes that these allegations have no structural safety significance.

   a. All of the grout in question survived the initial load application without failure (AC-26). The possibility of premature loading will be assessed as part of the overall programmatic review concerning procedures addressed under QA/QC Category 6 "QC Inspection."

   b. No infraction of installation procedures for anchor inserts was found (AC-31).

   c. The allegation that trash was dumped into the bottom of a concrete form cannot be substantiated. Even if true, the containment structure concrete, including the dome, was shown to be adequate and acceptable in the in-situ structural integrity test (AC-36).

   The TRT scheduled an interview with the alleger of AC-31 to discuss the TRT findings, but he declined to appear. A letter will be sent to him in lieu of a closing interview. The individual who made allegation AC-26 will be informed of the TRT's findings by letter. The TRT could not establish the identity of the individual who made allegation AC-36.

6. Actions Required: None.

1. Allegation Category: Civil and Structural 6, Rebar Improperly Installed or Omitted

2. Allegation Number: AC-30, AC-37, AQC-12, AC-38, AC-39 and AC-49

3. Characterization: It is alleged that reinforcing steel (rebar) was not properly inspected upon receipt at the site (AQC-12 and AC-37). It is also alleged that rebar was omitted in the following locations:

   a. A 6 foot x 6 foot section of concrete in the Safeguards Building (AC-30).

   b. The Unit 1 containment structure wall, specifically horizontal "tie" reinforcement (AC-38).

   c. Four column faces in the wall along column line EA of the Auxiliary Building (AC-39).

   d. It is also alleged that reinforcement was installed upside down in a building near the Unit 2 containment structure (AC-49).

   In addition to these allegations, the Region IV resident inspector requested that the NRC Technical Review Team (TRT) review the following possible reportable design deficiencies involving reinforcing steel (rebar):

   e. Reportable Design Deficiency Concerns:

      (1) Rebar was omitted in a reactor cavity concrete placement between the 812-foot and 819-foot, 1/2-inch elevations in the Unit 1 Reactor Building.

      (2) Brown & Root construction requested a change in the configuration of two rows by nine layers of No. 9 reinforcing bars (2 x 9 - #9), as shown on drawing 2323-S1-0572, Rev. 4, to a continuous circular arrangement.

      (3) Because of interferences with 14-inch diameter sleeves, the horizontal tails of No. 11 vertical reinforcing bars within the triangular columns surrounding the reactor cavity were modified to clear the sleeves. Also because of extreme congestion within the columns, stirrup details were modified.

      (4) Six No. 10 additional horizontal bars were omitted from a beam above a construction opening on column line KA between 6A and 7A in the Auxiliary Building.

      (5) Nine No. 9 and two No. 4 additional reinforcing dowels were omitted around an elevator shaft door in the Unit 1 Reactor Building.

      (6) Forty-six No. 9 dowels on the face of the wall in the excess letdown heat exchanger room in the Unit 1 containment structure were omitted.

(7) Ten No. 8 additional horizontal dowels were omitted from a beam over a construction opening in Safeguards Building No. 1.

(8) Brown & Root construction requested authorization to substitute No. 5 vertical wall rebars in lieu of the No. 8 wall rebars required in two corners of a wall in the Auxiliary Building.

4. <u>Assessment of Safety Significance</u>: The individual who made allegations concerning the improper receipt inspection of rebar (AQC-12 and AC-37) was not initially contacted by the TRT because the allegation was sufficiently clear to allow the TRT to proceed with its investigation. Allegations AC-38 and AC-39 concerning missing rebar in the Unit 1 containment wall and 4 columns in the Auxiliary Building were made by a former Brown & Root employee. The TRT attempted to determine this individual's identity, but could find no record of it. The TRT did not initially contact the individual who made the allegation concerning the rebar installed upside down (AC-49), because the alleger stated the problem had been corrected. The individual who made the allegation concerning the missing rebar in the Safeguards Building (AC-30) was contacted by the TRT to clarify his allegation.

The allegations that rebar was not properly inspected upon receipt (AQC-12 and AC-37) relate to the use of weldable reinforcing steel associated with the installation of radial shear-bar reinforcement at the base of the containment structure. At this location, Grade 60, 1-inch x 4-inch steel bars were joined by full penetration butt welds to No. 18 ASTM A615, Grade 60 reinforcing bars. Gibbs & Hill specification 2323-SS-10 required that a special chemical analysis be performed on each heat of reinforcing steel which was to be welded. Upon receipt, this reinforcing steel could be identified by the results of a special chemical analysis attached to the mill report. The QC inspector would verify that the results of the special chemical analysis conformed to the requirements of specification 2323-SS-10, and, if it was acceptable, QC personnel would then paint one end blue.

It is alleged the No.18 Grade 60 reinforcing steel was used prior to the proper inspection upon receipt by QC in 1975. The TRT reviewed testimony taken during an interview in which the alleger stated that the QC inspector was pressured into hurrying the inspection process and that the reinforcing steel that was used prior to QC inspection was subsequently inspected and accepted by the QC inspector. This may indicate a partial breakdown in the area of QC receipt inspection of reinforcing steel. The TRT reviewed the receipt inspection reports for all No. 18 reinforcing bars received in 1975 and determined that three shipments were received that had a special chemical analysis attached to the mill report. The receipt inspection reports for these three shipments were signed off by QC, indicating that an inspection had been performed. However, the TRT could not determine from its review of these receipt inspection reports whether any reinforcing steel was used prior to proper QC receipt inspection.

The TRT's safety assessment for the remaining allegations and reportable design deficiencies are discussed below:

a. During an interview with the alleger, the TRT learned that the allegation of missing rebar in the Safeguards Building actually referred to the return pump station at Squaw Creek Dam (AC-30). For the detailed assessment of this allegation, see Civil and Structural Category 12, AC-29.

b. This allegation (AC-38) was first reviewed in NRC Region IV Inspection Report No. 79-25, which refers to the omission of horizontal tie rebar in the Unit 1 containment structure, and concludes that the alleger was referring to an occurrence in the Unit 2 containment structure rather than in Unit 1. This event occurred shortly before the alleger terminated his employment, and it was assumed by the Region IV inspector to be the event to which he referred. The omission of horizontal shear tie reinforcement in Unit 2 was originally investigated in Region IV inspection report 79-18, which notes that this reinforcement had been omitted near the junction of the containment wall and the hemispherical dome and was subsequently placed at a higher elevation. An analysis by Gibbs & Hill (G&H) concluded that the structure would be capable of carrying the design loads with the reinforcement in the as-built location. To determine if the allegation did indeed pertain to the Unit 1 containment structure and if all the reinforcing steel was placed in the Unit 1 containment wall as required, the TRT reviewed all 33 concrete pour packages (101-5805-001 through 101-5805-033) pertaining to the main concrete placements in the Unit 1 containment wall. These pour packages contained rebar placement checklists which documented the results of inspections performed by B&R QC confirming the placement of the reinforcing bars to the applicable drawings. The TRT found three placement inspections in which the reinforcing bar placement was initially checked as unsatisfactory; the problems were then corrected, and the placement was signed off as satisfactory. The other 30 inspections performed were all checked as being satisfactory in that there were no deviations from the drawings.

c. On October 27, 1977, a nonconformance report (NCR) C-806 was issued reporting the omission of 12 No. 8 vertical wall reinforcing bars at 4 column locations in the wall along column line EA of the Auxiliary Building (AC-39). The reinforcing steel had been omitted between the 810-foot, 6-inch and 831-foot elevations and involved four separate concrete placements made from May to October 1977. This information was submitted to G&H engineering for resolution. G&H performed an analysis which showed that the columns remained capable of carrying the design loads without the missing reinforcing bars and further directed that the bars be omitted from the columns for the remainder of their height through the 873-foot, 6-inch elevation.

d. The TRT reviewed the April 10, 1979, transcript of a Region IV interview with an alleger and identified an allegation that reinforcement was installed upside down in a building near the Unit 2 containment structure (AC-49). However, during the interview the alleger claimed that the problem had been corrected prior to concrete placement.

e. (1) The reinforcing steel that was placed between the 812-foot and 819-foot, ½-inch elevations in the reactor cavity wall of the Unit 1 Reactor Building was completed and inspected to drawing 2323-S1-0572, Rev. 2. After the concrete was placed, Brown & Root received Rev. 3 to the drawing showing a substantial increase in reinforcing steel over that which was installed. G&H engineering was informed of the omission by Brown & Root nonconformance report C-669, which is referenced in the Brown & Root internal deficiency report CP-77-6. G&H engineering replied that the omission of this additional reinforcing steel did not in any way impair the structural integrity of the structure. G&H stated that the additional rebar was added as a precaution against cracking which might occur in the vicinity of the neutron detector slots should a loss of coolant accident (LOCA) occur. A portion of the omitted reinforcing steel was placed in the next concrete lift above the 819-foot, 1/2-inch elevation. G&H stated that this was done to partially compensate for the reinforcing steel omitted below and to minimize the overall area subject to possible cracking.

The TRT requested documentation to indicate that an analysis was performed supporting this conclusion. The TRT was subsequently informed that an analysis had not been performed.

(2) In response to Brown & Root construction's Request for Information or Clarification (RFIC) RBCR-37, Design Change/Design Deviation Authorization (DC/DDA) No. 832 was issued stating that the configuration of the 2x9-No. 9 reinforcing bars (two rows by nine layers), as shown on drawing 2323-S1-0572, Rev. 4, could be changed to a continuous circumferential arrangement. The TRT reviewed this drawing and determined that these bars were among those omitted in the concrete placement between the 812-foot and 819-foot, 1/2-inch elevations and subsequently placed above the 819-foot, 1/2-inch elevation (See e(1) above.) Revision 4 shows each of the four sets of No. 9 bars used to form the configuration required were to be bent in two places to form an approximate circular configuration when placed. The DC/DDA stated the bars could be bent to a specified radius to form a true circular arrangement. The change, therefore, only affected the way in which the bars were bent and did not reduce the load-carrying capacity of the structure.

(3) During the placement of reinforcing steel within the triangular columns surrounding the reactor cavity at the 826-foot, 11-inch elevation, interferences were encountered. The horizontal tails of the No. 11 vertical reinforcing bars were interfering with 14-inch-diameter sleeves already in place. The TRT reviewed DC/DDA No. 6918 and the attached sketches which showed that six bars were cut and replaced with bars tailed up to achieve total anchorage and three bars were bent down to clear the sleeves.

Also, due to congestion problems, the design of the No. 4 stirrups surrounding the ten No. 18 circular bars was modified to allow for installation. The stirrup design was modified to a two-piece design rather than one piece, as originally designed. This modification was permitted only within the triangular columns.

(4) On October 26, 1977, nonconformance report (NCR) No. C-809 was issued by Brown & Root reporting the omission of six No. 10 additional horizontal reinforcing bars from a beam over a construction opening on column line KA between 7A and 6A in the Auxiliary Building at the 831-foot, 6-inch elevation. G&H engineering issued DC/DDA No. 558 in response to the NCR. G&H engineering stated that the reinforcing bars were not required provided that one of the following conditions was met: (1) shoring remained within the construction opening until the slab above 831-foot, 6-inch elevation and the wall along column line KA above this elevation reached their design strengths, or (2) slab shoring remained adjacent to the construction opening until the concrete used to close the construction opening had reached its design strength. The intent was to provide adequate support to the 831-foot, 6-inch slab from either the wall above, the wall below, or from shoring. The disposition of the NCR showed that the shoring was left in the construction opening until the concrete wall and slab above had cured. The TRT reviewed the design change and solutions proposed and found the approach taken to be satisfactory. The TRT also reviewed drawing SAB-00711, which showed that the construction opening was closed with concrete pour No. 002-4810-042 on January 30, 1979.

(5) Brown & Root issued NCR No. C-810 reporting the omission of nine No. 9 and two No. 4 additional reinforcing dowels around the elevator shaft door in the Unit 1 Reactor Building at the 832-foot, 6-inch elevation. G&H DC/DDA No. 477 indicated that the nine No. 9 dowels were to be drilled and grouted in place, and that the two No. 4 dowels could be placed without doweling into the slab. A review of the safety implications of the omitted reinforcing bars by Texas Utilities Electric Company (TUEC) Design Engineering showed that cracking of the concrete in this area could have occurred during conditions such as a seismic event if the reinforcing steel had not been placed. The review concluded that the cracking would not have affected the safety of the structure.

(6) On October 31, 1977, NCR C-811 was issued by Brown & Root reporting the omission of 46 No. 9 dowels on the face of the wall in the Excess Letdown Heat Exchanger Room in the Unit 1 Reactor Building. The civil QC inspector involved stated that the reinforcing steel had been installed and checked but that it was subsequently removed to allow for the installation of steam generator lower supports and reactor coolant pump tie supports and not replaced. G&H engineering directed that the dowels be drilled and grouted in place.

(7) On October 21, 1977, concrete was placed which was to have contained ten No. 8 additional horizontal reinforcing dowels that were to run over the top of a construction opening in the Unit 1 Safeguards Building. NCR C-815 was issued by Brown & Root reporting this omission. In response to the NCR, G&H engineering decreased the size of the construction opening in the 7-S wall by placing a vertical construction joint 1 foot, 6 inches from the east face of the C-S/7-S column. Decreasing the size of the opening allowed the ten No. 8 reinforcing bars to be placed with sufficient anchorage length developed by hooking the ends down into the 1-foot, 6-inch space. The TRT reviewed drawing SSB-1065, which verified the decrease in opening size, and also showed the concrete pour numbers (105-4810-018 and 105-4810-034) for concrete placed in the 1-foot, 6-inch space and in the wall over the opening to the 829-foot, 6-inch elevation. A check of the rebar checklists included in these pour packages showed the rebar installation was inspected and accepted. Drawing SSB-1065 also showed that the construction opening was closed with concrete pour No. 105-4810-019.

(8) Brown & Root construction issued request for information or clarification (RFIC) C-1987 on November 3, 1977, which requested authorization to substitute No. 5 vertical reinforcing bars in the wall 5 feet, 4 inches north of column line 3-A for the widths of the column line F-A and G-A walls (corner bars) in lieu of the No. 8 bars shown on the drawings. This involves the intersection of two walls. In assessing this issue, the TRT reviewed drawing 2323-S-0751, Rev. 15, which showed that the vertical bars in one of the walls, 5 feet, 4 inches north of column line 3-A between F-A and G-A, are No. 8 at 8 inches center to center (8 @ 8") each face and that the horizontal reinforcing is No. 6 @ 8" each face. Drawing 2323-S-0746, which shows the other walls involved along column lines F-A and G-A north of column line 3-A to be secondary walls. Drawing 2323-S-0785 gives the reinforcing requirements for secondary walls when the reinforcing is not otherwise noted on the elevation drawing. The walls along column lines F-A and G-A north of 3-A are 1 foot thick and require No. 5 @ 8" each way in each face. The No. 5 bars as installed in the walls along column lines F-A and G-A are, therefore, acceptable. In summary, one wall had No. 6 and No. 8 bars and the intersecting walls properly had No. 5 bars. The question involves the correct bars to use at the point of intersection (corners). Drawing 2323-S-0785 also indicates that where two walls intersect, the types of vertical corner bars used should be based on the thicker and/or more heavily reinforced wall. The four bars required in each corner are No. 8 based on the reinforcing in the wall 5 feet, 4 inches north of column line 3-A. The TRT reviewed DC/DDA 518, Rev. 1, dated November 9, 1977, which also verified that the No. 5 bars were acceptable for the wall but that the No. 8 vertical wall bars were to be installed in the corner as required. The TRT also reviewed concrete pour package 002-4831-017, which showed that the

reinforcing steel installation as per DC/DDA 518 Rev. 1 was inspected and accepted and that the concrete was placed on November 11, 1977. Therefore, the TRT concluded that the correct bars were used.

The six documented structural sections with omitted reinforcing steel above indicate a breakdown in the quality control program as evidenced by the fact that these omissions were not detected prior to concrete placement.

5. Conclusion and Staff Positions: For the allegations concerning improperly inspected rebar (AQC-12, AC-37), the TRT concludes, based on the fact that the reinforcing steel used was subsequently accepted by QC, that this issue has no effect on the structural safety of the structure.

   The TRT reached the following conclusions for the remaining allegations and reportable design deficiencies:

   a. Allegation AC-30, which refers to the return pump station at Squaw Creek Dam, not the Safeguards Building, is examined in Civil and Structural Category 12, AC-29.

   b. For AC-38, the TRT concludes that the horizontal shear bar reinforcement was placed in the Unit 1 Containment Building wall as required and further agrees with the conclusion drawn in Region IV Inspection Report No. 79-25 that the allegation refers to the Unit 2 containment structure, where the G&H analysis showed that the structure would be capable of carrying the design loading with the reinforcing steel in its as-built location. Therefore, the TRT concludes that this issue has no structural safety significance.

   c. The TRT reviewed the G&H analysis and agrees with their methodology and conclusion (AC-39). The TRT, therefore, concludes that this allegation has no structural safety significance.

   d. The TRT concludes that since this instance of improperly installed rebar was corrected prior to concrete placement, this issue has no adverse effect on the structural safety of the structure.

   e. (1) The TRT cannot determine the safety significance of this issue until an analysis is performed verifying that the reinforcing steel in the as-built condition is adequate.

      (2) The TRT concludes that the change made to the No. 9 reinforcing bars did not affect the load-carrying capacity of the structure.

      (3) The TRT finds the modifications made to the interfering bars to be acceptable and to have no adverse effect on the structural safety of the structures.

      (4) The TRT finds that the omission of the additional reinforcing bars will have no adverse effect on the structural safety of the structure because shoring left in place until the concrete had cured made the additional reinforcing steel unnecessary.

(6) The TRT concludes, based on the fact that the reinforcing steel was subsequently placed as per the disposition of the NCR, that there is no adverse effect on the structural safety of the structure.

(6) The TRT concludes, based on the fact that the dowels were subsequently installed as per the disposition of the NCR, that this incident had no adverse effect on the structural safety of the structure.

(7) The TRT concludes that by decreasing the size of the construction opening, which allowed the reinforcing bars to be placed with sufficient anchorage length, this issue has no structural safety significance.

(8) Based on the fact that the No. 8 vertical wall bars were installed in the corners as required, the TRT concludes this issue has no structural safety significance.

However, the results of these evaluations which pertain to QC rebar placement and receipt inspection procedures will be further assessed as a part of the overall programmatic review concerning procedures addressed under QA/QC Category 6, "QC Inspection." Therefore, the final acceptability of these evaluations will be predicated on the satisfactory results of the programmatic review of this subject. Any adjustments to the existing conclusion of this evaluation resulting from the programmatic review will be reported in a supplement to this SSER.

Subsequent to its investigation, the TRT attempted to contact the individuals who made the allegations discussed above to inform them of the TRT's findings. The individual who made Allegations AQC-12 and AC-37 will be informed of the TRT's findings by letter. The individual who made Allegation AC-30 was informed of the TRT's findings by letter. The TRT has not been able to contact the alleger of AC-49.

6. Actions Required: TUEC shall provide an analysis of the as-built condition of the Unit 1 reactor cavity that verifies the adequacy of the reinforcing steel between the 812-foot and 819-foot, ½-inch elevations. The analysis shall consider all required load combinations.

1. Allegation Category: Civil and Structural 7, Uncontrolled Repair

2. Allegation Number: AC-10

3. Characterization: It is alleged that the removal of a Hilti bolt from the floor at the 852-foot level of the Safeguards Building resulted in a cone-shaped section of concrete being removed which was later repaired in an "uncontrolled manner."

4. Assessment of Safety Significance: The NRC Technical Review Team (TRT) did not initially attempt to contact the alleger because the allegation was sufficiently clear for the TRT to proceed with its investigation.

   In assessing this allegation, the TRT examined NRC Investigation Report 81-12 (April 16, 1982), which described the observations of the area in question by an NRC investigator and the senior resident inspector. They concluded that the floor was repaired with a surface patch rather than being repaired all the way through. Such an uncontrolled and undocumented repair of a portion of a Category I structure indicates a lack of QA/QC control.

   The TRT concurred with these findings based on its observations of the floor area in question. Nevertheless, the TRT performed an independent evaluation of the safety significance of a 14-inch-diameter hole extending through the floor slab adjacent to pipe support No. CC-1-137-700-E63R, as alleged. This hole is located in the Electrical and Control Building and not in the Safeguards Building, as alleged, and as reported in NRC Investigation Report 81-12.

   For the worst-case analysis, the TRT assumed that two reinforcing steel bars (rebars) were cut in the process of removing the Hilti bolt. To account for the unknown quality of the material used in the repair, the TRT computed the ultimate moment capacity of the floor slab with and without a 14-inch section of slab removed. These estimated strength capacities were compared to the strength requirements necessary to resist the actual moments resulting from the slab design loads. The adequacy of shear capacity was also verified in a similar manner. From these analyses, it was evident to the TRT that the slab in its as-built condition is capable of resisting the actual design loads, even though the most conservative engineering assumptions concerning cut rebar and a 14-inch hole were made.

5. Conclusion and Staff Position: Based on observations made by the TRT, the floor slab does not show any sign of degraded capacity or of poor repair practices. The slab appears continuous and composed of good materials. An independent TRT analysis of the slab capacity, based on conservative engineering assumptions, confirmed that the structural integrity of the slab would be maintained under its design loads. Accordingly, this allegation has no structural safety significance.

However, the lack of QC inspection will be further assessed as a part of the programmatic review concerning procedures addressed under QA/QC Category 6, "QC Inspection." Therefore, the final acceptability of this uncontrolled repair will be predicated on the satisfactory results of the programmatic review of this subject. Any adjustments to the existing conclusion of this evaluation resulting from the programmatic review will be reported in a supplement to this SSER.

The individual who made the allegation was contacted by the TRT to inform him of the TRT's finding. The alleger expressed his satisfaction with respect to the TRT's disposition of his allegation.

6. Actions Required: None.

1. Allegation Category: Civil and Structural 8, False/Wrong Documents

2. Allegation Number: AQC-1, AQC-2, AQC-3, AQC-7, AQC-46 and AQC-51

3. Characterization: It is alleged that the following records were falsified at various times:

   a. Concrete air entrainment records (AQC-1).

   b. Concrete laboratory test records (AQC-2). This allegation consisted of four separate parts: (1) that slump records were falsified, (2) that laboratory tests (air, slump, and temperature) for concrete placements of 10 cubic yards or less, prior to 1978, were not performed, (3) that laboratory tests were signed by a Level II inspector not present at the time the tests were performed, and (4) that the alleger signed a pressure gauge qualification test that he was not qualified to certify.

   c. Aggregate tests (January 1976). The alleger maintains that he and his foreman falsified these tests (AQC-3).

   d. Compression strength tests, at the direction of the general foreman and laboratory manager (AQC-7).

   e. Midpour tests during the placement of the Unit 1 Containment Building basemat on February 21, 1976 (AQC-46).

   f. Cadweld tensile test records were reported by an inspector without the tests actually being performed during the spring and summer of 1976 (AQC-51).

4. Assessment of Safety Significance: The NRC Technical Review Team (TRT) attempted to determine the identities of the individuals who made allegations AQC-1 and AQC-2 and could not find any record of their identities. Allegation AQC-46 appeared in a April 1979, Fort Worth Star-Telegram article; the allegation was made by three unidentified individuals. The TRT did not initially attempt to contact the individual who made allegations AQC-3, AQC-7, and AQC-51 because the allegations were sufficiently clear to allow the TRT to proceed with its investigation.

   a. In assessing this allegation (AQC-1), the TRT reviewed documents contained in Brown & Root (B&R) Deficiency and Disposition Report (DDR) No C-488 R1, R. W. Hunt Company QA Report HCP 21697 on concrete acceptance test results and the results of the Region IV investigation of this allegation (inspection report 77-02). The records showed that on January 20, 1977, a 3.9 percent air content value was recorded in the concrete acceptance test report (HCP 21697) as 4.3 percent by a Level I inspector. The incident was reported to R.W. Hunt management by a co-worker. R. W. Hunt then issued a DDR identifying the placement of out-of-specification concrete and corrected the air entrainment value in the acceptance test report. The Level I inspector was subsequently fired for his action.

To assess the possible safety significance of the falsification, the TRT examined the compressive strength test results for the concrete placement in question (105-7785-001) and found that the results ranged from 4905 psi through 5414 psi, and were well above the design specification strength of 4000 psi. The out-of-specification air content had little effect, if any, on the strength of the concrete placed.

b. (1) The TRT cannot determine if slump test results were or were not falsified based on an examination of test records (AQC-2). To assess whether the records, if falsified, could have adversely affected the strength of the concrete, the TRT reviewed the results of compression tests performed on the concrete placed between April 11 and 13, 1978. (These dates correspond to the dates of the alleged falsification.) The TRT found the compressive strength to be consistent with that of concrete placed before and after the dates and within the specification.

(2) The allegation that laboratory tests (air, slump, and temperature) were not performed on placements of concrete 10 cubic yards or smaller was investigated by the NRC Region IV staff (IE Inspection Report 78-07). This allegation was made in April 1978. The NRC Region IV staff reviewed log books that were the personal property of a number of laboratory personnel, but could not substantiate the allegation even though the alleger stated that such a review would be "revealing." The TRT reviewed all the concrete pour packages for the Pipe Tunnel, the Condensate Storage Tank and the Service Water Intake Structure, which are classified as safety related, to determine if any of the concrete placements were 10 cubic yards or less. The TRT identified eight concrete placements (111-1794-003, 111-1797-009, 111-1797-010, 111-1802-001, 111-9810-001, 035-9796-001, 035-9796-002, and 035-9796-003) that were placed prior to 1978 and that were 10 cubic yards or less. The dates of these placements were between August 1976 and February 1977. The eight concrete pour packages contained records showing field and laboratory tests results, but there was no way of determining whether the field tests were actually performed. However, for each of the above placements, concrete cylinders were also made and tested; the results demonstrated adequate strength. Prior to 1978, concrete placements in the Containment Structure, Fuel Handling Building, Auxiliary Building and Safeguards Building were generally for structural elements such as walls, slabs, and foundations. The placements for these types of elements would generally be 50 yd^3 or larger. To identify placements of 10 yd^3 or less the TRT identified nonconformance reports concerning repair work (voids, honeycomb, etc.) to walls, slabs, etc. because placements for repair work would generally be less than 10 yd^3. The TRT identified four concrete placements which needed concrete repair, but were repaired by means of "dry-pack" or "grout." The TRT interviewed four former R. W. Hunt employees who were involved in concrete testing activities at Comanche Peak during the time period in question. These four employees were employed on site with

another employer. Three of them were working in the concrete testing laboratory. All four stated that they did not participate in or observe any falsification and/or failure to perform required concrete tests.

(3) The allegation that a Level II inspector signed reports for tests performed on September 3 and 4, 1977, that he could have had no knowledge of was also reviewed by NRC Region IV personnel (IE Inspection Report 78-07). The alleger stated he had obtained this information from another individual who thought the falsification occurred in December 1977. The Region IV inspectors reviewed the daily payroll records of all laboratory personnel for the first 10 days of September and all of December 1977. The Level II inspector was present every day in September, but was absent December 4, 5, 11 and 18 through 31. The Region IV staff could find no reports validated by the Level II inspector for the days alleged. The TRT reviewed the Region IV inspection report and concurred with the approach taken and the results of the investigation. In addition to reviewing the Region IV inspection report, the TRT examined strength test results for the concrete placed during the period stated in the allegation and found them to be above the minimum required design strength.

(4) The allegation that the alleger signed a pressure certification test that he was not qualified to certify (on August 15, 1977) was investigated by NRC Region IV personnel (IE Inspection Report 78-07). Through an interview with Brown & Root calibration facility personnel, the NRC Region IV investigator learned that the pressure gauge was calibrated by Brown & Root personnel in accordance with their procedures. The calibration record was an R. W. Hunt form signed by the alleger who observed the test in accordance with the R. W. Hunt procedure. Prior to the Region IV investigation, B&R issued a DDR (February 17, 1978) that described this situation as a pre-existing and continuing problem in general, and proposed corrective action. The NRC Region IV staff concluded that while the allegation was substantiated, there were no safety consequences since the calibration was performed by a qualified individual in accordance with prescribed procedures. The TRT reviewed the Region IV inspection report, and concurred with the approach taken and the results of the investigation.

c. The allegation (AQC-3) was first evaluated by the NRC Region IV staff (IE Inspection Report 79-09). The alleger, a former R. W. Hunt employee, stated that the falsification by him and his "foreman" occurred during the first 3 or 4 weeks of his employment, beginning January 19, 1976. The NRC Region IV staff reviewed the prequalification tests performed by Texas Industries, the aggregate supplier, on the material supplied to the site between January and May 1976 and also examined the results of in-process concrete testing. Both sets of results complied with the specification requirements. The NRC Region IV staff also determined through discussions with a TUEC representative that the "foreman" was a Level

II inspector in charge of the work. The NRC Region IV staff concluded that any falsification of test results on the part of the alleger would not have had a significant adverse impact on the quality of the concrete. The testing performed by Texas Industries was for the purpose of material qualification, whereas the tests performed by R. W. Hunt Company were to monitor the material for any deviation from the specification and to assure concrete of uniform workability and strength. The tests performed to verify concrete workability and strength were the test for slump and the cylinder test for compressive strength. The Texas Industries tests and the tests on fresh concrete indicated that the aggregate was satisfactory for its intended purpose.

In addition to reviewing the Region IV report, the TRT examined the results of slump and compressive strength tests for the period in which the falsification was alleged to have occurred. The test results were within specified limits and were consistent with concrete produced before and after this period.

d. The 2 individuals making the allegation (AQC-7) and 13 other individuals were questioned by Region IV personnel between April 5, 1979 and May 7, 1979, regarding the allegation (IE Inspection Report 79-09). One of the allegers denied the allegation, stating he was misquoted in the newspaper. Another stated that he did not falsify concrete records himself but knew of other inspectors who had. One of the other 13 individuals interviewed stated he thought that falsification occurred, but did not know when or by whom. In addition, the NRC Region IV staff examined the test result statistics of the concrete produced prior to and during the period of the alleged falsification and did not find any apparent variation in the uniformity of the concrete. The NRC Region IV staff concluded that the allegation could not be substantiated. The TRT staff reviewed slump and air entrainment test results of concrete placed during the period the remaining alleger was employed (January 1976 to February 1977) and did not find any apparent variation in the uniformity of the parameters for fresh concrete placed during this period. However, since air content and slump tests have been alleged to be falsified, the TRT believes that additional action is required by TUEC to confirm that the results of the strength tests are representative of the strength of the concrete placed.

e. According to an article that appeared in the Fort Worth Star-Telegram (April 1979), three unidentified R. W. Hunt Company concrete inspectors alleged that during the placement of 6600 cubic yards of concrete for the Unit 1 Containment Building basemat on February 21, 1976, some concrete was not tested, but instead the results were written in as averages (AQC-46). The concrete specification in force at this time required that slump, air content, temperature, and cylinders be taken every 100 cubic yards. The TRT reviewed concrete pour package (101-2805-001) for this placement and found 67 sets of test cylinders with the associated results of slump, air content, and temperature as per the specification. However, the TRT cannot determine from a review of these records whether the field tests were actually performed. Since the results of compression tests performed on the concrete cylinders would be the final measure of its acceptability, the TRT reviewed these results and found them to be acceptable and within the specification.

f. The TRT reviewed all 440 Cadweld tensile test results for 1976 and identified 30 tests that were performed by the inspector in question (AQC-51). Twenty-eight of these tests were performed on one single day (October 13, 1976), while the other two tests were performed on two different days (July 21, 1976 and August 20, 1976). The TRT cannot determine whether or not all of the 30 tests in question were performed or if results were falsified and did not specifically look into the falsification issue. The remaining 410 tests performed by other inspectors all met the tensile strength requirements. The 30 Cadwelds tested were removed from the first layer of the exterior wall of the Unit 1 Containment Building at the 832-foot, 6-inch elevation. The TRT reviewed tensile test results of other Cadwelds performed by the individuals who made the Cadwelds in question. The results were found to be satisfactory. The Cadweld rejection rate for the 21 Cadwelders who made the 440 Cadwelds ranged from zero percent to four percent, with one at six percent.

The fact that the allegations concerning the falsification of an air entrainment test and the certification of the pressure gauge test were substantiated indicates a partial breakdown in the QA/QC program in these areas.

5. Conclusions and Staff Position: The allegation (AQC-1) that a concrete air entrainment record was falsified is true. Even so, the compressive strength of the concrete in question was within specifications.

The allegation (AQC-2) that slump tests on April 11 and 13, 1978, were performed incorrectly and that the results were falsified could well be true and cannot be refuted. The TRT examined the compressive strength test results of the concrete in question and found that they were within specifications.

The allegation (AQC-2) that laboratory tests for small placements were falsified was found to have no structural safety significance since, in addition to the recorded laboratory tests, the validity of which was questioned, cylinder strength tests were also performed to demonstrate adequate strength. In addition, in interviews with the TRT, former employees of the R. W. Hunt Co., who worked during the time period cited in the allegations, denied the validity of the allegation. Furthermore, the limited number of concrete placements of less than 10 cubic yards, even if improperly tested, would have little structural safety significance.

The allegation (AQC-2) that an inspector signed test results for which he could have had no knowledge could not be substantiated because no reports could be found which had been signed by the inspector on the days alleged. Even if the allegation were true, test results showed the strength of the concrete placed during the period of the allegation to be above the minimum required strength.

The allegation (AQC-2) that the alleger signed a pressure gauge test which he was not qualified to certify was found to have no structural safety significance since the alleger did not actually perform the calibration.

The TRT cannot determine the validity of the allegation (ACQ-3) that concrete aggregate tests were falsified. Nevertheless, the concrete placed during the period cited in the allegation was consistent with that of concrete placed before and after this period.

The validity of the allegation (AQC-46) that midpour tests were falsified during the placement of the Unit 1 Containment Building basemat cannot be determined. The results of compression tests indicate that the concrete placed was of high quality.

The TRT cannot determine the validity of the allegation (AQC-51) that Cadweld tensile test results were falsified. If this falsification did indeed occur, the structural integrity of the exterior wall of the Unit 1 Containment Building has not been violated because (1) the tensile strength of other Cadweld test specimens performed by the 21 Cadwelders were found to be satisfactory, (2) the Cadweld rejection rate for each Cadwelder is at an acceptable level, (3) and the containment structure met all the criteria for displacement and cracking control as well as structural rebound when subjected to 115 percent of design pressure, as stated in CPDA-31, 792, "Final Report on Structural Integrity Test for Unit 1 Concrete Containment Structure."

Accordingly, the above allegations have no structural safety significance. However, the allegations resolved on the basis of acceptable concrete strength test results may need to be further assessed pending the resolution of allegation AQC-7. Also, the results of these evaluations pertaining to QC inspection procedures will be further assessed as a part of the overall programmatic review concerning procedures addressed under QA/QC Category 3, "Records." Therefore, the final acceptability of these evaluations will be predicated on the satisfactory results of the programmatic review and the satisfactory resolution of allegation AQC-7. Any adjustments to the existing conclusions of these evaluations will be reported in a supplement to this SSER.

The allegation (AQC-7) that compressive strength test results were falsified cannot be closed at this time. The TRT agrees with the Region IV staff that the uniformity of the fresh concrete placed during this period suggests that there was no serious problem with the hardened concrete and, therefore, no serious safety problem. However, this conclusion is based on air content, slump, and strength tests, all of which have been alleged to be falsified. The issues regarding air content and slump, as well as other allegations discussed above, were resolved on the basis of the concrete strength test results. Due to the importance of the concrete strength test results, the TRT concludes that additional action by TUEC is necessary to provide confirmatory evidence that the reported concrete strength test results are indeed representative of the strength of the concrete placed in the Category I structures.

The TRT is attempting to contact the individual who made allegations AQC-3, AQC-7, and AQC-51 to inform him of the TRT's findings.

6. Actions Required: TUEC shall determine areas where safety-related concrete was placed between January 1976 and February 1977, and provide a program to assure acceptable concrete strength. The program shall include tests such as the use of random Schmidt hammer tests on the concrete in areas where safety is critical. The program shall include a comparison of the results with the results of tests performed on concrete of the same design strength in areas where the strength of the concrete is not questioned, to determine if any significant variance in strength occurs. TUEC shall submit the program for performing these tests to the NRC for review and approval prior to performing the tests.

1. Allegation Category: Civil and Structural 9, QC Inspector Training

2. Allegation Number: AQC-9

3. Characterization: It is alleged by two former R. W. Hunt Company employees that (a) after a March 1977, NRC investigation, closed book recertification tests of R. W. Hunt inspectors were done "open book" and that (b) tests were given with the answers provided.

4. Assessment of Safety Significance: The NRC Technical Review Team (TRT) did not initially attempt to contact the two allegers because the TRT was able, with some initial investigation, to clarify the allegations sufficiently to proceed with its evaluation.

   The allegation about the recertification tests refers to recertification testing that was required because a Region IV investigation in 1977 (inspection report 50-445/77-02) questioned the most recent certification of R. W. Hunt Level I and II inspectors. The NRC Region IV staff found that R. W. Hunt did not comply with the minimum 2-year experience requirement for qualification as a Level I concrete inspector, as required by the ASME Code to which they were committed by the Preliminary Safety Analysis Report (PSAR). The R. W. Hunt Skills Training Certification manual stated that "Experience requirements may be reduced if the individual can demonstrate capability in a given job through previous performance or satisfactory completion of an examination and orientation training." Also, the Region IV staff found that the "Certification of Qualifications," which was issued to each Level I and II inspector did not include the activity the inspector was qualified to perform or the basis used for certification, as was required. Each candidate for certification was required to demonstrate proficiency in performing specific practical tests on one or more samples approved by the Level III examiner. The Region IV staff found that R. W. Hunt had permitted a Level I inspector to perform concrete cylinder compression tests and aggregate sieve analysis without evidence of demonstrated proficiency and approval in accordance with the above requirements.

   As a result of this investigation, Brown & Root (B&R) audited R. W. Hunt training and certification activities and required each inspector to be recertified by attending specific training sessions and by closed book testing. The work performed by the personnel qualified under the previous provisions was reviewed by B&R QA personnel and found to be within the specification requirements. In addition B&R assigned a QC civil engineer to work full-time with R. W. Hunt on site to ensure full compliance with project requirements.

   Between April 3, 1979, and May 7, 1979, NRC Region IV inspectors interviewed 15 individuals associated with concrete testing activities regarding the allegation concerning the recertification tests (inspection report 50-445/79-09), including the 2 who had originally made the newspaper allegations (April 1979). The alleger who stated in the newspaper article that after the NRC investigation of March 1977, the recertification of inspectors to test cylinders was "open-book" did not mention open-book testing when interviewed by the Region IV inspectors. He stated that he had failed a Level I soils test, that he was subsequently given answers (orally) by the laboratory manager, and then repeated the soils test using the notes he had taken. He also stated that to obtain his recertification (after March 1977) he needed only to have a supervisor sign the recertification form.

When interviewed by Region IV, the other individual who alleged in the newspaper that he had been given answers to tests reaffirmed his allegation. To determine if the individual was referring to the recertification tests or to tests he had taken prior to March 1977 to obtain his initial certification, the TRT reviewed his employment records. The TRT found that this individual was not employed by R. W. Hunt at the time the March 1977 recertification tests were given. His allegation therefore would be referring to tests he took prior to March 1977 to obtain his initial certification. Two other individuals who were questioned by the Region IV staff between April 3, 1979 and May 7, 1979 generally supported the allegations. Again, to determine which tests these individuals were referring to, the TRT reviewed their employment records and the results of the Region IV interviews. One individual states that the recertification tests were administered properly; the other was not employed by R. W. Hunt Co. at the time the recertification tests were given. Therefore, these two individuals would be referring to tests they had taken prior to March 1977, their initial certification tests. These three individual's certifications were among those questioned by the 1977 Region IV investigation (inspection report 77-02) and their work had been audited and found acceptable.

In summary, the allegation that recertification tests were administered "open-book" was supported only by the individual making the allegation. The allegation that test answers were given for tests taken prior to March 1977 was supported by three individuals. Eleven other individuals who were questioned did not support the allegations.

The TRT reviewed the personnel file of the individual who made the allegation concerning the open book recertification tests and learned that he was employed from August 16, 1976, to June 28, 1978. Therefore, he was among those inspectors whose previous work had been reviewed and found acceptable. The TRT also found copies of tests taken by the individual for recertification in concrete and soil inspections. The TRT reviewed the test the individual had taken relating to concrete cylinder tests and could not determine whether the test had been administered properly. The TRT also examined test result statistics for concrete placed from 1975 to 1978, and found that the concrete placed was of uniform quality and strength and that there was no apparent variance in the test results. Approximately 35 different inspectors were involved in concrete testing between 1975 and 1978; more than 7 inspectors conducted concrete compression tests on a rotating basis during this period.

5. Conclusion and Staff Position: The allegation that answers to tests were given prior to March 1977 cannot be refuted. An NRC Region IV investigation (Inspection Report 77-02) questioned the qualifications of the R. W. Hunt inspectors. The work performed by the R. W. Hunt inspectors

certified prior to March 1977 was reviewed by B&R and was found to be within specifications, a fact subsequently reported to NRC Region IV staff. Therefore, the TRT concludes that this allegation has no structural safety significance.

The allegation that the recertification tests for concrete cylinder testing were given "open book" cannot be substantiated. This allegation was not supported by any of the other individuals questioned, which suggests it was an isolated occurrence. The work performed by this individual prior to March 1977 was audited and found to be satisfactory, which would indicate the individual possessed the knowledge required to properly perform the required testing. In addition, the test results for concrete placed, including the concrete compression tests, were contributed to by many inspectors whose qualifications were acceptable. These test results showed the concrete was of uniform quality and strength. Based on the fact that the inspector's work had previously been reviewed and found to be acceptable, and that a number of inspectors contributed to the test results, which showed the concrete to be of uniform quality, the TRT concludes that this issue has no structural safety significance.

The results of these evaluations will be further assessed as a part of the overall programmatic review concerning inspector qualifications addressed under QA/QC Category 4, "Training and Qualification of Personnel." Therefore, the final acceptability of these evaluations will be predicated on the satisfactory results of the programmatic review of this subject. Any adjustments to the existing conclusion of this evaluation resulting from the programmatic review will be reported in a supplement to this SSER.

The TRT contacted one of the allegers who made one of the allegations discussed above. He declined to meet with the TRT. He will be informed by letter of the TRT's findings. The TRT is attempting to contact the other alleger involved.

6. Actions Required: None.

1. Allegation Category: Civil and Structural 10, Improper Testing

2. Allegation Number: AQC-4, AQC-5, AQC-6, AQC-8, AQC-11 and AQC-48

3. Characterization: It is alleged that the following violation of testing procedures occurred:

   a. Equipment required for aggregate testing was sitting unused on laboratory shelves (AQC-4).

   b. Shortcuts were taken on tests involving grading of aggregate (AQC-5).

   c. During the placing of a 6600-cubic-yard section of the basemat for Unit 1, some concrete was placed without the required testing (AQC-6).

   d. Concrete cylinder compression tests were run at a faster loading rate than permitted by NRC regulations (AQC-8).

   e. Concrete test cylinders with adequate strength were used to represent other placements (AQC-11).

   f. Concrete test cylinders in the Hunt Laboratory moist room were allowed to dry (AQC-48).

   Allegations AQC-4, AQC-5, AQC-6, and AQC-8 were investigated by NRC Region IV and documented in inspection report 79-09, which was reviewed by the NRC Technical Review Team (TRT) as a step in its own assessment of the allegations.

4. Assessment of Safety Significance: Allegations AQC-4, AQC-11, and AQC-48 were judged as having sufficient clarity for technical resolution without initial contact between the TRT and the alleger. Allegations AQC-5, AQC-6, and AQC-8 were made in newspaper articles which did not identify the allegers, and the TRT has been unable to determine the allegers' identities.

   a. The test equipment that allegedly remained unused at the project laboratory was for the test for Potential Reactivity of Aggregates (Chemical Method), American Society for Testing and Materials (ASTM) C 289. Test Laboratory Manual TLM-004 (CP-QP-0.5), which was in effect during most of the construction period, required that the test be run once for each 4000 tons of aggregate. The TRT inspected "Folder 1 - Potential Reactivity, 4000 Ton Test" and learned that between May 6, 1975, and July 12, 1978, there were 60 tests for potential reactivity. The TRT also interviewed the laboratory technician who performed most of the tests. This period covers the bulk of heavy construction and the entire employment period of the alleger. The testing rate during this period exceeded one test per 4000 tons of coarse aggregate. Thus, testing was at a higher rate than required by the testing requirements.

b. The shortcut alleged is that TUEC used a hot plate for drying aggregate in its sieve analyses of coarse aggregate rather than an oven, as specified in test method ASTM C-136. Note 4 of that method contains the following information:

> Samples may be dried at the higher temperatures associated with the use of hot plates without affecting results, provided steam escapes without generating pressures sufficient to fracture the particles, and temperatures are not so great as to cause chemical breakdown of the aggregate.

The alleged shortcut, then, is permitted by the provision just cited.

c. The TRT inspected all batch tickets and test records for the 6600-cubic-yard basemat placement and physically inspected those portions of the mat still accessible. A placement that size required 66 sets of test cylinders, with associated data on fresh concrete. There were 67 sets of records in the file, all of which showed compliance with specifications. While little of the placement was available for inspection, that portion that could be seen was in excellent condition. The implied aspect of falsification is dealt with in Civil and Structural Category 8.

d. Cylinder strength testing must be done in accordance with "Method for Compressive Strength of Cylindrical Concrete Specimens," ASTM C-39. That method permits any rate of loading during the first half of the loading range, but restricts the rate of loading at fracture to the range of 20 to 50 psi per second. A higher rate of loading may produce a higher indicated strength. The definitive work on investigating the effect of rate of loading on indicated strength (Watstein, D., "Effect of Straining Rate on the Compressive Strength and Elastic Properties of Concrete," Proceedings, American Concrete Institute, Vol. 49, 1953, p. 729) demonstrated a significant increase in indicated strength for very high dynamic rates of loading. However, a rate 100 times that specified produces an indicated increase in strength of only 10 percent. The testing machine used to break cylinders on the Comanche Peak project, a Forney Model CAC-50-DR, if run at maximum capacity, could achieve a testing rate no greater than 20 times the specified rate. This rate could produce an apparent increase in strength of about 6.5 percent. For 4,000 psi concrete the apparent increase would be about 250 psi. In a detailed check by the TRT of several placement packages, and a spot check of others, the 4000 psi concrete averaged more than 5000 psi, and individual results exceeded 4500 psi. Thus, if some tests were conducted at too high a loading rate, no results were changed from failing to passing. If there were tests within 6.5 percent of the design strength not detected by the TRT, the strength could be expected to gain 6.5 percent within a few weeks, so that the design strength would be attained long before the structure was put into service.

e. The TRT investigated the number of cylinders available for switching to other placements. The alleger stated that the switch occurred after "a good sample was found." By the time the 28-day tests were completed, at most, two extra cylinders remained for which the test results could be switched to the testing data for other placements. Unless this was a widespread practice, its significance would be small because of the relatively few cylinders available. The allegation does, however, raise a question as to the effectiveness of quality control in the laboratory.

f. To investigate the allegation that concrete cylinders in the laboratory moist room were allowed to dry, the TRT examined the current procedure for documenting moist room conditions and interviewed a Level II inspector who was present throughout the period when the laboratory was operated by the R. W. Hunt Company. At present there is a thermometer which provides a permanent temperature record. While there is no quantitative measurement of humidity, there are daily visual observations of the presence or absence of fog in the room. These observations are also a part of the record.

During the R. W. Hunt operation, temperatures were recorded, but there apparently was no documentation of humidity. Batch plant inspectors were required to note the condition of the moist room, but there is no record of their observations. There is a history of breakdowns in the water supply to the laboratory, and a shutdown as long as 6 hours has been documented. With the door to the moist room closed, there would be a negligible drop in humidity during such a period. As long as the relative humidity remains above 90 percent, concrete curing conditions are favorable. It is pertinent to note that any drying that might occur would produce conservative results in that measured strengths would be lower than actual strengths.

5. Conclusion and Staff Positions: The TRT concludes that these allegations have no impact on structural safety.

a. All required tests for ASTM C-289 were performed.

b. The alleged shortcut in carrying out aggregate grading tests is permitted by the provisions of the specified test method in ASTM C-136.

c. All required testing was carried out in connection with the 6600-cubic-yard basemat placement.

d. Although this allegation may have been true, the fastest possible loading of test cylinders would have increased the indicated strengths by no more than 6.5 percent and would have had no effect on the acceptability of the concrete.

e. The alleged substitution of test cylinders is unlikely to have affected a sufficient number of cylinders to have had a material effect on the overall test results.

f. Although the allegation that the laboratory failed to maintain the water supply at all times may be true in that there were brief shutoffs of water to the moist room humidifiers, these periodic breakdowns would result in conservative strength results on concrete cylinders.

Accordingly, these allegations have no structural safety significance. However, the effectiveness of quality control in the laboratory will be further assessed as part of the overall programmatic review concerning procedures addressed under QA/QC Category 6, "QC Inspection." Therefore, the final acceptability of this evaluation will be predicated on the satisfactory results of the programmatic review of this subject. Any adjustments to the existing conclusion of this evaluation resulting from the programmatic review will be reported in a supplement to this SSER.

The alleger for allegations AQC-4 and AQC-48 cannot be located. Allegations AQC-5, AQC-6, and AQC-8 were made in newspaper articles in which the allegers were not identified. The TRT has been unable to determine their identities. The TRT sent a letter to the alleger of AQC-11 explaining its disposition.

6. <u>Actions Required</u>: None.

1. <u>Allegation Category</u>: Civil and Structural 11, Seismic Design/Construction

2. <u>Allegation Number</u>: AC-41

3. <u>Characterization</u>: It is alleged that there was poor workmanship regarding the use of elastic joint filler material ("rotofoam") as a temporary spacer during construction to maintain the required air space between seismic Category I structures.

4. <u>Assessment of Safety Significance</u>: This allegation was received anonymously; therefore, the TRT could not contact the alleger about its evaluation of AC-41.

   TUEC informed NRC Region IV on November 23, 1977, of this allegation, which TUEC received anonymously in a telephone call on November 22, 1977. A Region IV inspector reviewed the allegation during an inspection conducted between November 28 and December 2, 1977, and concluded, based on the information available to him at the time, that all temporary rotofoam had been removed from the seismic gap between Category I structures. The matter was left open pending a Region IV review of the Brown & Root (B&R) QA/QC inspection and documentation program, which was being initiated to assure that the required seismic gap between Category I structures was being maintained. Rotofoam was used as a temporary spacer during construction to maintain this gap. Once the concrete hardened, the rotofoam was removed to eliminate any load transfer or dynamic interaction between buildings. If the relative motion between buildings was small and the presence of rotofoam was considered in the dynamic analysis of the building, leaving the rotofoam in place may not have had a significant impact on the dynamic performance of the buildings.

   During an inspection between January 3 and 13, 1978, the Region IV inspector reviewed B&R procedure CP-QCI-2.4-9, "Inspection of Elastic Joint Filler Material Removal," Revision 1 (December 12, 1977), and B&R inspection reports for December 15, 1977, and January 3, 1978, and had no further questions regarding this matter.

   The NRC Technical Review Team (TRT) attempted to obtain a further clarification of the concerns expressed by the alleger; however, neither TUEC nor the Region IV office had records of the alleger's telephone conversation other than what is stated above. The TRT determined, however, that prior to the time the allegation was made there was a misunderstanding as to whether or not the rotofoam should remain in place as part of the final construction. A letter from Gibbs & Hill (G&H) of September 6, 1977 (GTT-1543), indicated that construction was improperly proceeding on the basis that the rotofoam could be left in place. The letter further stated that this assumption was not in accordance with the facility design drawings and design concept and that expansion joints above grade should consist of a clear gap between buildings, i.e., free of rotofoam. As noted in the G&H letter, it was intended that the rotofoam be left in place below grade. Since construction had proceeded above grade, TUEC instructed B&R, in a letter of October 7, 1977 (TUS-5012), to remove the rotofoam above grade. As noted, B&R procedure CP-QCI-2.4-9 was also implemented to verify removal of the rotofoam. Based on discussions with TUEC and G&H engineers, the TRT found that the rotofoam was to be left in place for the expansion joints above grade between the Safeguards Building and the Reactor Building.

If properly implemented, B&R procedure CP-QCI-2.4-9 should have provided an adequate inspection record for demonstrating that the air gap between buildings was adequately maintained. However, the TRT found only two inspection reports relating to this procedure (the December 15, 1977 and January 3, 1978, reports referenced). These reports did not fulfill the complete inspection requirements of CP-QCI-2.4-9. Furthermore, this procedure was deleted on July 18, 1978 (B&R memo IM-14835). A G&H memo of January 30, 1978 (GHF-2390) indicated that an inspection was made on November 23, 1977, and stated that the removal of the rotofoam from the subject areas was acceptable. However, the memo related only to construction at that point and did not provide any documented evidence of the inspections that were made.

A B&R interoffice memo of February 19, 1978 (IM-12934), discussed an inspection of the seismic gap between the Auxiliary Building and the Containment Building for Unit 1. The memo indicated that the removal of rotofoam was not completed and requested further removal and/or engineering evaluation. TUEC engineers apparently investigated this matter; however, the TRT found no formal documentation indicating the resolution of this matter.

Between September 14, 1978, and October 17, 1978, a B&R QC inspector made additional inspections of the air gap between seismic Category I structures. Six different areas were inspected. In five out of the six areas, the inspector indicated unsatisfactory conditions due to the presence of foreign material in the air gap, such as wood wedges, rocks, clumps of concrete, and rotofoam. These unsatisfactory inspection reports were officially resolved on April 18, 1983, in response to NCR C-83-01067 (April 13, 1983). The disposition of this NCR noted that "field investigation reveals that most of the material has been removed." Based on discussions with TUEC engineers, it is the TRT's understanding that field investigations were made but that no permanent records of these investigations were maintained. TUEC engineers provided the TRT with five pages of field measurements made between March 15 and March 24, 1983, which indicated that investigations of the air gap between the Auxiliary Building and the Fuel Building were conducted. These measurements appeared to indicate that the required air gap was not provided to the 813-foot, 6-inch elevation (the required elevation in procedure CP-QCI-2.4-9). Even though the measurements indicated a nonconforming condition, TUEC could not provide any documentation indicating whether an engineering analysis was performed to justify this nonconformance or whether the material was subsequently removed. The TRT attempted to inspect the air gap between the structures but could not because in most cases the final joint sealer or roof flashing had already been installed. In several areas between the Auxiliary Building and the Safeguards Building, the air gap could be observed and appeared to be clear of any obstructions. In one doorway between the Safeguards Building for Unit 1 and the Auxiliary Building at the 830-foot, 6-inch elevation, the air gap was clear to an observer looking up. However, a wooden board and other debris were observed when viewed straight in and downward.

5. Conclusion and Staff Positions: Based on the review of available inspection reports and related documents, on field observations, and on discussions with TUEC engineers, the TRT cannot determine whether an adequate air gap has been provided between concrete structures. Field investigations by B&R QC inspectors indicated unsatisfactory conditions due to the presence of debris in the air gap, such as wood wedges, rocks, clumps of concrete and rotofoam. The disposition of the NCR relating to this matter states that the "field investigation reveals that most of the material has been removed." However, the TRT cannot determine from this report (NCR C-83-01067) the extent and location of the debris remaining between the structures.

   Based on discussions with TUEC engineers, it is the TRT's understanding that field investigations were made but that no permanent records were maintained. In addition, it is not apparent that the permanent installation of elastic joint filler material ("rotofoam") between the Safeguards Building and the Reactor Building, and below grade for the other concrete structures, is consistent with the seismic analysis assumptions and dynamic models used to analyze the buildings, as these analyses are delineated in the Final Safety Analysis Report (FSAR). The TRT, therefore, concludes that TUEC has not adequately demonstrated compliance with FSAR Sections 3.8.1.1.1, 3.8.4.5.1, and 3.7.B.2.8, which require separation of seismic Category I buildings to prevent seismic interaction during an earthquake.

   Depending on the extent of nonconformance with FSAR Sections 3.8.1.1.1 3.8.4.5.1, and 3.7.B.2.8, the allegation is judged to have merit and potential safety significance. Prompt remedial actions as delineated below should be implemented.

   No closing interview could be held regarding this allegation, because the allegation was received anonymously.

6. Actions Required: TUEC shall:

   1. Perform an inspection of the as-built condition to confirm that adequate separation for all seismic Category I structures has been provided.

   2. Provide the results of analyses which demonstrate that the presence of rotofoam and other debris between all concrete structures (as determined by inspections of the as-built conditions) does not result in any significant increase in seismic response or alter the dynamic response characteristics of the Category I structures, components, and piping when compared with the results of the original analyses.

1. Allegation Category: Civil and Structural 12, Concrete Construction and Deficiencies/Tolerances

2. Allegation Number: AC-29

3. Characterization: It is alleged that a spillway pillar, span, or column was erected 75 degrees to 80 degrees offset and that reinforcing steel was omitted from a concrete wall.

4. Assessment of Safety Significance: There are two spillways at Comanche Peak Steam Electric Station (CPSES). One, the service water discharge spillway, is located near the safe shutdown impoundment (SSI); the other is located at the Squaw Creek Dam. The alleger stated that the construction in question took place some time between 1976 and 1977. The spillway at Squaw Creek Dam was constructed between August 1976 and January 1977, so it was considered to be the spillway in question. The spillway at Squaw Creek Dam, however, does not have a span, column or pillar. Therefore, on August 3, 1984, the NRC Technical Review Team (TRT) interviewed the alleger to clarify this allegation.

From the interview, the TRT learned that the spillway pillar, span, or column to which the alleger referred was located in the Service Outlet Structure below the Squaw Creek Dam Spillway, which does have a suspended structure and supports that could be described as a span and pillars.

The TRT inspected the service outlet structure at the Squaw Creek Dam Spillway and found no evidence of any spillway pillar, span, or column which was erected 75 to 80 degrees offset. The TRT also determined that the general configuration of the structure was consistent with that shown on the following drawings:

FN-SCR-37	FN-SCR-48
FN-SCR-39	FN-SCR-49
FN-SCR-40	FN-SCR-71
FN-SCR-42	FN-SCR-72
FN-SCR-44	

The TRT learned during an interview with the alleger that the allegation concerning the 6-foot by 6-foot concrete wall area of the Safeguards Building, which allegedly had no reinforcement placed around a pipe approximately 24-inches wide, was incorrect. (Refer to Civil and Structural Category 6, Allegation AC-30.) The alleger identified the 6-foot by 6-foot concrete wall area as located in a structure near the Squaw Creek Dam Spillway.

The TRT inspected the structures located near the Squaw Creek Dam spillway and found two structures with a 2- to 3-foot-diameter pipe surrounded by reinforced concrete. One of these was the outlet works conduit section; the other was the return pump station. The TRT examined 141 concrete placement cards associated with these two structures.

The TRT determined that the conduit section was placed between June 27 and November 17, 1975, and the return pump station section was placed between March 31, 1976 and February 10, 1977. Because the alleger's employment on the project began in 1976, the TRT concluded that the allegation, if valid, concerned the return pump station. There are two 24-inch steel pipes in the return pump station which pass through a concrete wall. The TRT reviewed reinforcement drawings (FN-PS-35 and FN-PS-36) for the wall at the return pump station and found that the wall section surrounding the pipe was designed to have the following reinforcement:

a. Eight No. 5 diagonal bars at the inside face
b. Eight No. 7 diagonal bars at the outside face
c. Ten No. 7 vertical bars at the outside face
d. Ten No. 5 vertical bars at the inside face
e. Ten No. 7 dowels (lap spliced with item c)
f. Eight No. 5 dowels (lap spliced with item d)

The walls of the return pump station were placed on June 21, 1976. The TRT examined the pertinent concrete placement card. It contained the required two signatures certifying that the reinforcement was correctly placed prior to concrete placement.

5. Conclusions and Staff Position: Since the structures at which the alleged construction deficiencies occurred are categorized as nonsafety related (FSAR Volume IV Section 3.2), the allegation is judged by the TRT to have no safety significance. Furthermore, the TRT concludes that the first part of the allegation is not valid because a structure that was constructed at 75 degrees to 80 degrees offset from the intended geometry could not be accepted by inspection personnel without detection of such a significant deviation. Field inspection by the TRT indicates correct alignment.

The TRT further concludes that the second part of the allegation is not valid because the concrete placement card indicates that the reinforcement was placed.

Additional evidence is provided by the fact that the portion of the wall surrounding the 24-inch pipes has been subjected to the maximum static load stress for which it was designed. The soil pressure has been in place and acting upon the wall for several years, and the reservoir was completely filled by water pumped through two 24-inch diameter pipes passing through the wall; therefore, this portion of the wall has also been subjected to whatever vibratory loads may be imparted to the wall by the pumping operation. Inspection by the TRT revealed no distress in the wall, and the structural integrity of the wall was observed to be intact.

Accordingly, this allegation has neither safety significance nor generic implications.

The TRT informed the alleger by letter of its disposition of this allegation.

6. Actions Required: None.

1. Allegation Category: Civil and Structural 13, Cracks in Concrete Beneath the Reactor Vessel

2. Allegation Number: AC-44

3. Characterization: It is alleged that detrimental cracks exist in the concrete pad at the bottom of the reactor vessel.

4. Assessment of Safety Significance: The NRC Technical Review Team (TRT) did not initially attempt to contact the alleger because the allegation was sufficiently clear to the TRT to proceed with the investigation.

   The existence of these cracks is documented in Nonconformance Report (NCR) C-650. The cracks are in a lift of concrete near the bottom of the reactor vessel, but not in the basemat. The TRT examined concrete placement package 101-2812-001 and NCR C-650, and found no documented violations of specifications during the concrete placement, which occurred on March 21, 1977. Subsequent to placement, however, vertical cracks occurred that extended horizontally to the edge of the reactor cavity. The Gibbs & Hill (G&H) design engineer stated on May 11, 1977, that he found the cracks during his investigation. He attributed the cracks to the mass, configuration, and formwork on the interior circumferential face, all of which precluded normal shrinkage, and stated that the cracks were of no structural significance.

   An NRC Region IV structural engineer also presented his evaluation of the cracks during the ASLB hearing conducted on June 9, 1982. The TRT reviewed this testimony, along with that of G&H engineers given on June 7 and and 8, 1982, and agreed with the assessment contained in their testimony. The doughnut shape of the concrete section and the rigid form in the opening made it virtually impossible to avoid cracking if the entire section was placed in one pour, as it was. However, the structure was adequately reinforced so that the cracks would not impair structural behavior and capacity. The cracks have been repaired at the surface with epoxy resin for operational, rather than structural, reasons. The TRT inspected the concrete and found it to be in excellent condition.

   The TRT review of the design indicated that the concrete section was originally designed as two sections, with construction joints at the locations where the cracks occurred. The contractor was given the option of placing the concrete either in two sections with construction joints, or in one section without joints. The cracks that formed were not greatly different from the construction joint which would have been present if the two-placement option had been adopted, thus, the concrete in place essentially conformed with the original design.

   One of the cracks was near the mid-span of a deep beam spanning a 20-foot cavity. Reinforced concrete beams must crack in the bottom tensile zone when load is applied. If flexural stresses were kept below the tensile strength of concrete, less than 20 percent of the strength of the steel would be utilized. In design, the reinforcement is distributed so that the cracks are numerous and very narrow, both for the sake of appearance and to prevent corrosion of the steel. The occurrence of a pre-existing crack merely changes the distribution of cracks; the total width of the cracks in the tensile zone remains unchanged. A crack in the upper compressive stress zone closes when load is applied and is rendered innocuous.

The beam section must also be capable of carrying shear stresses. The cracks observed should not produce a critical situation because shear stresses are low near midspan and because crack planes are normally irregular so that aggregate interlock, particularly in the tightly closed compressive zone, resists shear stress. The biggest defense against shear, however, is the fact that the concrete was heavily over-reinforced. The critical load condition is not the static load condition, nor even the earthquake condition, but the differential pressure resulting from a postulated accident condition. For this condition all the load is carried by the steel, with no credit given to the concrete, and the presence of cracks in the concrete is immaterial. The design of the section was controlled by thickness requirements for shielding. The section was thicker and, therefore, stronger than required to carry the loads. The cracks did not make the steel vulnerable to corrosion because the upper surface, which provides the most likely ingress for water, is sealed, and the bottom surface is in a dry environment.

5. Conclusion and Staff Positions: Although the allegation is correct in citing the existence of cracks, it is not correct in imputing detrimental structural consequences to them. The safety of the structure is not adversely affected by the cracks. Accordingly, this allegation has neither safety significance nor generic implications.

   The TRT held a closing interview with the alleger, who was satisfied with the TRT's disposition of this allegation.

6. Actions Required: None.

1. Allegation Category: Civil and Structural 14, Control Room Area Deficiencies

2. Allegation Number: AE-17

3. Characterization: It is alleged that the field run conduit, the drywall, and the lighting installed in the area above the ceiling panels in the control room are classified as non-seismic and are supported only by wires and that these items may fall as a result of a seismic event.

4. Assessment of Allegation: The NRC Technical Review Team (TRT) did not initially attempt to contact the alleger because the allegation was sufficiently clear to allow the TRT to proceed with its investigation.

   The TRT electrical group reviewed the electrical aspects of this allegation. (See Electrical and Instrumentation Category 4.) The Civil and Mechanical group of the TRT evaluated the seismic aspects of this allegation.

   General Design Criteria No. 19 requires that safe occupancy of the control room during abnormal conditions be provided for in its design. The Comanche Peak Steam Electric Station (CPSES) control room is in a seismic Category I structure, with certain seismic Category II and nonseismic components located in the ceiling. Seismic Category I refers to those systems or components which must remain functional in the event of an earthquake. Seismic Category II refers to those systems or components whose continued functioning is not required, but whose failure could reduce the functioning of any seismic Category I system or component (as defined in Regulatory Guide 1.29) to an unacceptable level or could result in an incapacitating injury to occupants of the control room. Seismic Category II systems or components are, therefore, designed and constructed so that a Safe Shutdown Earthquake (SSE) will not cause such failure or injury.

   In assessing this allegation, the TRT reviewed the CPSES nonsafety-related conduit, lighting fixtures, and the suspended ceilings installed in the control room. Three types of suspended ceiling exist in the control room: drywall, louvered, and acoustical. The following list designates those ceiling elements present in the control room and their seismic category designation:

1.	Heating, Ventilating and Air Conditioning	- Seismic Category I
2.	Safety-related Conduits	- Seismic Category I
3.	Nonsafety-related Conduits	- Seismic Category II
4.	Lighting Fixtures	- Seismic Category II
5.	Sloping Suspended Drywall Ceiling	- Nonseismic
6.	Acoustical Suspended Ceiling	- Nonseismic
7.	Louvered Suspended Ceiling	- Nonseismic

   The TRT also examined the control room ceiling system and pertinent design drawings and met with cognizant Texas Utilities Electric Company (TUEC) engineers on July 31, 1984, to discuss the specific seismic analyses performed for the ceiling elements. In addition, the TRT held a conference call on August 1, 1984, with principal Gibbs & Hill (G&H) design engineers (at which TUEC representatives were present) to discuss the design and calculation procedures for the ceiling elements.

The TRT determined that none of the suspended ceiling elements were considered to be either seismic Category I or II; however, TUEC had modified the sloping suspended drywall to add more support. G&H could not provide backup calculations to support this modification, nor could TUEC provide justification for their position that the remaining suspended ceiling elements (i.e., the louvered and acoustic elements) would not fall and cause an incapacitating injury to operating personnel. This would indicate failure of the quality assurance program to ensure that applicable provisions of Regulatory Guide 1.29 were fully met.

The TRT requested backup calculations for the sloping suspended drywall. TUEC provided the calculations on August 3, 1984, along with the calculation packages for the lighting fixtures, the nonsafety-related conduits larger than 2 inches in diameter, and the safety-related conduit. The TRT reviewed these calculations, except those for the safety-related conduit, since they were designated as seismic Category I and therefore were excluded from the scope of this review.

The TRT found that nonsafety-related conduits that were less than or equal to 2 inches in diameter were not supported by redundant seismic Category II cable restraints. The TRT also verified the adequacy of calculations for the nonsafety-related conduits larger than 2 inches in diameter.

The TRT found that the G&H calculations were based on the equivalent static load method, which involves multiplication of the dead weight of an item by an appropriate seismic acceleration coefficient. This equivalent static load calculation did not take into account the influence from the adjoining suspended ceilings on the calculated response. This was significant because redundant cable supports were not provided for the suspended louvered and acoustical ceilings, and the impact from the accelerations of the lighting fixtures was not considered in any analysis. The ceiling, as a whole, manifested a more complex configuration than that assumed in the equivalent static load analysis in that the effects from adjoining suspended ceilings were not considered. A justification based on the seismic response characteristics of the entire ceiling, which would account for the frequency content and amplification characteristics of the seismic motions, as represented by floor response spectra, is required to justify the value of the seismic acceleration coefficient used.

5. Conclusions and Staff Position: The TRT found that not all items in the Control Room ceiling fall under the seismic Category I or II designation. Specifically, these items are the suspended drywall, acoustical, and louvered ceilings. These components, designated as nonseismic, do not satisfy the provisions of Regulatory Guide 1.29, since they were not designed to accommodate seismic effects. Nonsafety-related conduits that are 2 inches in diameter and less also were not designed to accommodate seismic effects. TUEC presented no evidence which showed that the effect of failure of these items had been considered.

The TRT concludes that calculations supporting the seismic Category II lighting fixtures do not adequately reflect the rotational interaction with the nonseismic items. In addition, the fundamental frequencies of the supported masses were not calculated to determine the influence of the seismic response spectrum at the control room ceiling elevation.

The individual who made the allegation discussed above will be contacted by the TRT upon resolution of this issue to inform him of the action taken.

6. Actions Required: TUEC shall provide:

   1. The results of seismic analysis which demonstrate that the nonseismic items in the control room (other than the sloping suspended drywall ceiling) satisfy the provisions of Regulatory Guide 1.29 and FSAR Section 3.7B.2.8.

   2. An evaluation of seismic design adequacy of support systems for the lighting fixtures (seismic Category II) and the suspended drywall ceiling (nonseismic item with modification) which accounts for pertinent floor response characteristics of the systems.

   3. Verification that those items in the control room ceiling not installed in accordance with the requirements of Regulatory Guide 1.29 satisfy applicable design requirements.

   4. The results of an analysis that justify the adequacy of the nonsafety-related conduit support system in the control room for conduit whose diameter is 2 inches or less.

   5. The results of an analysis which demonstrate that the foregoing problems are not applicable to other Category II and nonseismic structures, systems, and components elsewhere in the plant.

1. Allegation Category: Civil and Structural 15, Rebar Improperly Drilled

2. Allegation Number: AC-13, AC-14, AC-15, AC-18 and AC-40

3. Characterization: It is alleged that undocumented and unauthorized holes were drilled through reinforcing steel (rebar). The issue includes allegations relating to:

   a. the loan of rebar drills without proper documentation (AC-13),

   b. the unauthorized cutting of rebar in non-specific locations (AC-14, AC-18, AC-40), and

   c. the unauthorized cutting of rebar used in the installation of the trolley process aisle rails in the Fuel Handling Building (AC-15).

4. Assessment of Safety Significance: The NRC Technical Review Team (TRT) contacted the individual who made allegations AC-13 and AC-14 to clarify his concerns. The TRT did not initially attempt to contact the individuals who made allegations AC-18, AC-40, and AC-15.

   a. AC-13 concerns the loan of rebar drills allegedly used for the unauthorized cutting of rebar. During the NRC investigation of this matter, the NRC Office of Investigation (OI) interviewed nine individuals alleged to have knowledge of unauthorized cutting of rebar. These individuals provided sworn statements denying any knowledge of this activity. These statements are a part of OI Report A4-83-005 (May 20, 1983), which concludes that "there was no testimony received indicating that holes were drilled or rebar was cut without proper documentation, and no evidence was found to contradict the testimony of these individuals." One instance of possible unauthorized cutting of rebar is discussed in a supplement to the OI report (September 7, 1983). This instance is discussed below in relation to allegation AC-15.

   Because the alleger did not specifically identify who made unauthorized cuts of rebar, or where this cutting took place, the TRT attempted to quantify the amount of rebar that allegedly was cut without authorization. In discussions with the TRT, the alleger estimated that approximately five percent of the diamond core drill bits ordered by him were used in an unauthorized manner. He further estimated that one drill could be used to cut up to five rebars, depending upon the extent of cutting required. Although he could not be specific as to how many drills he ordered, the alleger thought that the number would be in the thousands. The NRC Region IV Investigation of this issue indicated that 415 diamond core drill bits were purchased during the period in question (IE Report 83-27). Using the actual number of drill bits purchased, together with the information provided by the alleger, the TRT estimated that there could be approximately 100 alleged unauthorized rebar cuts. Considering the large amount of reinforcing steel used in the plant, and the fact that the structures consist primarily of heavily reinforced concrete walls and slabs, the TRT determined that, if such unauthorized rebar cutting occurred, the amount involved would have an inconsequential effect on the safety of the structures.

b. Allegations AC-14, AC-18 and AC-40 also raise questions regarding the unauthorized cutting of rebar, but do not identify specific locations. During the course of the NRC Region IV investigation of this matter, the alleger provided a log book which, it was reported, would identify the unauthorized and undocumented rebar cutting. However, the Region IV inspector could not identify one rebar cut listed in the log that was not authorized. The TRT also reviewed the log and came to the same conclusion.

In discussing this matter with the TRT, the alleger confirmed that there was documentation supporting "ninety nine and three quarter percent" of the rebar cuts identified in the log. As part of Report 83-27, the NRC Region IV investigator traced 32 authorizations, approximately half of the documents noted in the log for the rebar cutting. He found that in all cases rebar cuts were properly identified on a design change authorization (DCA) or on a component modification card (CMC). In addition, the rebar cuts were traced to and identified on specific building structural drawings, with the corresponding authorizing document number. The TRT reviewed 10 CMCs and confirmed the findings of the Region IV investigation.

In reviewing authorizations in the log, the TRT noted that certain CMCs involved a number of rebar cuts in one area, and selected these for review. In one case, 7 different CMCs (3307, 3664, 3665, 3666, 3667, 3668 and 3669) seemed to pertain to one area and accounted for 68 rebar cuts. Upon reviewing the documentation, the TRT found that these cuts were made in a tunnel area in the Fuel Building. (The alleger identified this as a location where a large number of rebar was cut.) However, the 68 cuts were arranged such that only 9 bars actually had been cut. In another case, the log indicated 25 rebar cuts pertaining to CMC 00979. In this case, the TRT determined that all the cuts were made on one reinforcing bar in a support beam. Finally, the log indicated eight rebar cuts pertaining to CMC 3022. Once again, these eight cuts were to one bar in a support beam. All cuts were made in accordance with the rebar cutting criteria provided by Gibbs & Hill. These examples also illustrate the point that a large number of rebar cuts recorded are not necessarily synonymous with an identical number of rebar actually being cut. In all cases, one bar was cut a number of times, but adjacent bars were not. Thus, the cuts were arranged to minimize the overall effect on the strength of the structure.

The TRT estimates that approximately 335 rebar cuts are indicated in the alleger's log. Discussions with the alleger revealed that he believes he cut approximately five percent more rebar than was authorized, a number that corresponds to approximately 17 unauthorized rebar cuts. As noted earlier, such a number would have little effect on the safety of the structures.

c. Allegation AC-15 identifies a specific instance of the possible unauthorized cutting of rebar. In this case, a former Brown & Root employee stated he possibly drilled holes through rebar in a concrete floor without a component modification card (CMC) or a design change authorization (DCA). He explained that in January 1983 he drilled approximately 10 holes about 9 inches deep while installing 22 metal plates with a core drill. He said the metal plates were used to secure the trolley process aisle rails located on the 810-foot, 6-inch floor level in Room 252 of the Fuel Handling Building.

The TRT inspected the trolley process aisle rails and its anchoring system and observed no violations of project drawings or specifications. The TRT reviewed the reinforcement drawings (2323-S-0800 and 2323-S-0820) for the Fuel Handling Building to determine the location of rebar. The drawing showed three layers of reinforcement in the upper part of the mat, which consisted of a No. 18 bar running in the east-west direction, in the first and third layers, and a No. 11 bar running in the north-south direction, in the second layer.

The review of the reinforcement drawings (2323-S-0800 and 2323-S-0820) revealed that the layout of the east-west reinforcement and the trolley process aisle rails was such that only one bar of the east-west reinforcement could be cut by drilling holes for rail anchors. However, if 9-inch holes were drilled, both layers of the No. 18 reinforcing bar would be cut. Design Change Authorization (DCA) No. 7401 was written for authorization to cut the uppermost No. 18 bar at only one rail, but it did not reference the authorization to cut the lowermost No. 18 bar. The DCA (No. 7041) also stated that the expansion bolts and baseplates could be moved in the east-west direction to avoid interference with the No. 11 reinforcement running in the north-south direction. The information described in DCA No. 7041 was substantiated by Gibbs & Hill calculations. The DCA approval was based on the understanding that only the uppermost No. 18 reinforcement would be cut. If the 10 holes were actually drilled 9 inches deep, then the allegation that reinforcement was cut without proper authorization may be valid.

The DCA indicated that the holes were drilled to accommodate 1/2-inch Hilti bolts, which require a minimum embedment of 5-1/2 inches (as noted in Fig. 39, Sh. 5 of 5, attached to DCA-7041). Since there was no need to drill the holes deeper than 5-1/2 inches, the alleger may not be correct in stating that the holes were drilled 9 inches deep.

Although the allegations discussed above, with the exception of AC-15 which requires further action, cannot be substantiated, the fact that such allegations were made indicated that there was no effective quality assurance program to oversee the issuance and use of diamond core drill bits.

The TRT interviewed the individual concerned about the loan of rebar drills without proper documentation and the unauthorized cutting of rebar at nonspecific locations to inform him of the TRT's finding. This individual did not agree with certain TRT findings. In particular, the alleger felt that the TRT's estimate of approximately 120 unauthorized rebar cuts was much too low. He believes that the number of drill bits ordered by him was in the thousands and that as much as 20 percent of the drill bits may have been used in an unauthorized manner. It was also his opinion that the unauthorized cutting of rebar was not limited to his period of employment, but occurred for the duration of the project.

As a result of these additional discussions with the alleger, the TRT searched TUEC's files relating to the purchase of diamond drill bits and found that 1170 drill bits were purchased between January 13, 1978 and January 14, 1980. This number is more in agreement with the alleger's assessment and is higher than the previously reported number of 415 (IE Report 83-27). The TRT also found that there were a total of 3368 drill bits ordered from one manufacturer between January 13, 1978 and March 18, 1983. After this period, other manufacturers supplied the drill bits. Based on the usage through March 10, 1983, the TRT estimates that approximately 5000 diamond drill bits have been used to date on the project. Assuming that 20 percent of these drill bits were used in an unauthorized manner and that each drill bit could be used to cut up to five rebars, the TRT estimates that there could be approximately 5000 alleged unauthorized rebar cuts.

The TRT estimated that, depending upon the average length of rebar assumed, there are approximately 800,000 to 1,200,000 bars installed in all of the concrete structures. Thus, if 5000 bars were cut without authorization, they would represent approximately 0.6% of the total rebar in the plant. Even if all 5000 drill bits were used in an unauthorized manner it still would only represent 3% of the total rebar in the plant. Thus the percentage of rebar that could have been cut without proper authorization is low. Since no information was supplied to the contrary, the TRT assumed that these unauthorized cuts, if they did occur, were scattered throughout the plant and not concentrated in one localized area. In addition, as noted earlier, a large number of rebar cuts are not necessarily synonymous with an identical number of rebar actually being cut. It is also noted that nuclear structures are very conservatively designed. In addition to the conservative loads, load combinations, and safety factors utilized in the design, it is the common practice of the design engineer to specify 5 to 10 percent more rebar than is actually required by his calculations. This occurs because it is difficult to obtain the exact area of reinforcement required using standard bar sizes and standard bar spacing. The area of reinforcement is selected from charts which show the area provided for each bar size at a given spacing. Rather than underdesigning, the designer selects an area of reinforcement from the charts which is higher than that which is actually required. In addition, because critical structures contain a large number of bars, they are not generally vulnerable to the random cutting of a small number of bars.

5. Conclusion and Staff Positions: The TRT concludes that allegations AC-13, AC-14, AC-18 and AC-40 have no structural safety significance.

a. The allegations were not specific as to who made unauthorized cuts of rebar or where the cuts took place.

b. The number of unauthorized rebar cuts alleged, if true, would have an inconsequential effect on the safety of the structures.

However, the results of these evaluations will be further assessed as a part of the programmatic review concerning procedures addressed under QA/QC Category 6, "QC Inspection." Therefore, the final acceptability of these evaluations will be predicated on the satisfactory results of the programmatic review of this subject. Any adjustments to the existing conclusion of this evaluation resulting from the programmatic review will be reported in a supplement to this SSER.

Allegation AC-15 will remain open until the information requested of Texas Utilities Electric Company (TUEC) in "Actions Required" is provided.

The TRT attempted to contact the individuals who made allegations AC-18 and AC-15 to inform them of the TRT's findings. The individual who made allegation AC-18 will be informed of the TRT's findings by letter. One of the two individuals involved in allegation AC-15 cannot be located; the TRT is still attempting to contact the other. The TRT also contacted the individual who made allegations AC-13, AC-14, and AC-40 to discuss the TRT's findings pertaining to the concerns he raised in the first closure interview. An interview was arranged; however, later the alleger indicated he did not want to meet with the TRT. A letter will be sent to him informing him of the TRT's findings.

6. <u>Actions Required</u>: TUEC shall provide:

   1. Information to demonstrate that only the No. 18 reinforcing steel in the first layer was cut, or

   2. Design calculations to demonstrate that structural integrity is maintained if the No. 18 reinforcing steel on both the first and third layers was cut.

1. Allegation Category: Civil and Structural 16, Excavation and Backfill

2. Allegation Number: AQ-64

3. Characterization: It is alleged that overexcavation and improper fill under the Unit 1 Containment Building could invalidate the expected seismic response of the foundation due to the change in properties resulting from the removal of in-situ material.

4. Assessment of Safety Significance: The NRC Technical Review Team (TRT) did not initially attempt to contact the alleger because the allegation was sufficiently clear to allow the TRT to proceed with its investigation.

   During an investigation conducted in 1984, the NRC Office of Investigation (OI) interviewed the alleger (84-006, 3/7/84, A-7) and reference was made to overexcavation and improper repairs in the foundation rock for the Unit 1 Containment Building. The alleger stated that the excavation was erroneously made 6 to 8 feet too deep and that upon realization of the error, the repair technique was simply to throw the loose rock back into the excavation and fill it in with concrete.

   The TRT reviewed NRC inspection reports, the FSAR, and the Atomic Safety and Licensing Board (ASLB) hearing transcript, where this concern was the subject of contention No. 7 and was admitted into the hearing on June 16, 1980.

   By order of March 5, 1982, the ASLB granted summary disposition of contention No. 7, based on the finding that no genuine issue as to any material fact was shown by any of the filings. The TRT also reviewed the affidavits and statements filed by TUEC and by the NRC in support of the motion for summary disposition. These documents adequately describe rock overbreak, accompanying fissures, and subsequent repairs. Affected areas were backfilled with concrete having a minimum compressive strength of 2,500 pounds per square inch at 28 days, or were grouted to maintain continuity of the competent rock in which fissures were identified. The TRT reviewed the procedures utilized to replace fractured rock with dental concrete and to grout surrounding fissures and the accompanying compressive test results. The TRT found that FSAR figures 2.5.4-33a through 2.5.4-35 are maps of the excavation showing the location of fractures and the extent of dental concrete backfill. These figures showed that the area of overexcavation represented a small portion of the entire excavated area. FSAR figure 2.5.4-37, sheets 1 through 21, showed photographs of the excavated walls. The TRT interviewed the NRC inspector who was present during the excavation process and verified the conditions presented in the FSAR.

   The TRT independently evaluated the potential impact on the seismic response of the Unit 1 containment foundation due to the replacement of a limited amount of original rock with dental concrete from the standpoint of possible changes in foundation stiffness. Because of the facts that (a) the dental concrete's behavior, stiffness, and structural strength were essentially identical to those of the natural rock replaced at the site as indicated by the foundation report and (b) the area affected by the replacement work was relatively small (refer to FSAR Figures 2.5.4-33a through 2.5.4-35), the TRT determined that no appreciable impact on either the static or dynamic response characteristics of the foundation resulted from the overexcavation. An evaluation prepared by a geotechnical engineer in the NRC's Office of Nuclear Reactor Regulation supports this conclusion. He evaluated the effects on static and dynamic foundation stability of replacing undisturbed limestone and claystone foundation rock with dental concrete and concluded that the ability of the repaired foundation materials to withstand seismic disturbances had not been impaired.

5. Conclusion and Staff Positions: The TRT concludes that the overexcavation of a small portion of the Unit 1 Containment Building foundation and the subsequent replacement of the affected area with 2500 psi strength dental concrete and grout did not affect either the static or dynamic characteristics of the foundation. Therefore, the expected seismic response has not been invalidated as alleged. The excavation and repairs have had no safety impact upon foundation integrity. Accordingly, this allegation has neither safety significance nor generic implications.

   The TRT has contacted the alleger to arrange an interview to inform him of the TRT's finding.

6. Actions Required: None.

1. Allegation Category: Civil and Structural 17, Concrete Sampling

2. Allegation Number: AQC-45

3. Characterization: It is alleged that personnel produced incorrect readings on concrete batch plant scales by leaning on the wires connecting the weighing hoppers to the scales.

4. Assessment of Safety Significance: The NRC Technical Review Team (TRT) did not initially attempt to contact the alleger because the allegation was sufficiently clear to allow the TRT to proceed with its investigation.

The allegation refers to "some type of sampling machine that tells whether there are good samples or bad samples in the concrete." The information is attributed to a friend of the alleger's who was an equipment operator at the concrete batch plant. NRC Inspection Reports 50-445/83-01 and 50/446/83-03 clarified the allegation, indicating that the equipment operator is reported to have "charged that some personnel biased the operation of the concrete batch plant scales by leaning on the wires connecting the scales to the sensors."

During the major construction phase of the project, the concrete plant consisted of two identical batching systems, one feeding a mixer drum and one batching directly into ready-mix trucks. Although the system with the self-contained mixer was removed prior to the TRT review, the other system was still in use at the time of the TRT review. The batching system included three mechanical lever dial scales, each controlled by a wire connected to the weighing hopper with which it is associated. Each wire entered the scale room over a pulley at the top of the room and ran downward vertically about 6 feet to the loading lever of the scale. The scale dials faced the control room, from which they were visible through a large window. An electronic sensor connected to the scale provided a digital readout of each scale reading in the control room. It was possible to decrease the scale reading by entering the scale room and horizontally deflecting the vertical wires.

The TRT interviewed the concrete plant maintenance man who was present during most of the construction and who, at the time of the review, was also serving part time as operator because of the small amount of concrete production. He stated that the scale room was enclosed, well-illuminated, and provided with a large window so that all parts of the enclosure were visible from the control room, making it easier to prevent surreptitious tampering with the scale mechanisms. Thus, it would be obvious to everyone in the control room and to many outside the control room if anyone opened the scale room door, entered, and deflected the wires. The maintenance man was not aware that such an incident had ever occurred.

Since it was not possible to rule out such tampering completely, the TRT investigated the potential consequences of such tampering. During a concrete placement, a member of the TRT entered the scale room when all hoppers were loaded and deflected each scale wire as far as could be conveniently done by one person. The scale readings were affected as follows:

	Normal Reading	Deflected Reading
Aggregate scale	14,500 lbs	14,400 lbs
Cement scale	3,150	3,050
Water scale	1,040	990

The manner in which the scales are constructed makes possible only a decrease in readings, not an increase, if the scales were tampered with as alleged. This arrangement rules out the most common allegation of fraud in concrete batch plants: inflation of the cement batch weight. If the cement scales were tampered with, it would be necessary to add extra cement to satisfy the stipulated batch weight. Moreover, the possible change in aggregate weight is within permitted tolerances and may be ignored. The only ingredient of concern is water. Tampering with the water scale could cause an extra five percent to be batched, which would increase the water-cement ratio from 0.50 to 0.525. However, water is the one ingredient in concrete whose abuse in batching is most easily detected in fresh concrete. A five percent increase of water would increase the slump by nearly 2 inches. The good control of slump, as verified by test data in the many concrete placement packages reviewed by the TRT, strongly indicated that there was no tampering with the water scale, the only scale vulnerable to the type of tampering alleged which would adversely affect safety. Thus, the evidence suggests either that tampering did not occur or that it was confined to scales where tampering would have either no effect or a beneficial effect on the concrete.

5. Conclusion and Staff Positions: The TRT interviewed relevant personnel, observed the layout of the scale room, conducted a demonstration of the tampering alleged, reviewed test data on freshly placed concrete, and examined two NRC inspection reports in evaluating this allegation. Based on its findings, the TRT concludes that the allegation can be neither verified nor refuted. However, if tampering did occur, it was confined to scales where tampering would have either no effect or a beneficial effect on the concrete. Accordingly, this allegation has no structural safety significance. However, the results of this evaluation pertaining to QC controls at the batch plant will be further assessed as part of the overall programmatic review concerning procedures addressed under QA/QC Category 6, "QC Inspection." Therefore, the final acceptability of this evaluation will be predicated on the satisfactory results of the programmatic review on this subject. Any adjustments to the existing conclusion of this evaluation resulting from the programmatic review will be reported in a supplement to this SSER.

   The individual who made this allegation will be informed of the TRT's findings by letter.

6. Actions Required: None.

1. Allegation Category: Miscellaneous 1, Nuclear Fuel

2. Allegation Number: AM-2

3. Characterization: It is alleged that nuclear fuel was received prior to issuance of a special nuclear material (SNM) license.

4. Assessment of Safety Significance: The NRC Region IV (RIV) staff received this allegation by telephone in January 1983, from a member of the public. The caller had friends working at Comanche Peak Steam Electric Station (CPSES) who claimed that fuel was received onsite before the NRC issued an SNM license. Despite requests from RIV for either more specific information about the allegation or the identity of those making the allegation, the NRC staff received no substantive information because the caller stated that the allegers feared they would lose their jobs.

   The NRC Technical Review Team (TRT) reviewed the SNM license for Texas Utilities Electric Company (TUEC), issued February 14, 1983, the "dummy fuel assembly" receipt package, and the first-fuel receipt package. During this review, the TRT found that TUEC received a "dummy fuel assembly" onsite on December 15, 1982, at 9:30 a.m., as noted on Form RFO-201-1, "Fuel Receiving Record - Shipment Report." Form RFO-201-1 also shows that the first-fuel shipment was received onsite on May 4, 1983, at 1:45 a.m.

   The TRT interviewed both the Radiation Protection Engineer (RPE) and the Radiation Protection Supervisor (RPS) who stated that receipt of the "dummy fuel assembly" was used as a training exercise to prepare for incoming shipments of fuel expected to arrive at CPSES. The RPS stated that when CPSES received the "dummy fuel assembly," fuel-receipt procedures were followed as if it were a "real case" shipment. Thus, it is conceivable that the alleger mistakenly assumed that the "dummy fuel assembly," which was received onsite prior to issuance of the SNM license, was actually nuclear fuel.

5. Conclusion and Staff Positions: Based on a review of documents and forms, on interviews with the RPE and RPS, and on a review of an NRC Region IV interoffice memorandum assessing TUEC's program for receiving fuel, the TRT concludes that TUEC has an adequate program to receive fuel and did not actually receive fuel onsite prior to issuance of an SNM license on February 14, 1983. Accordingly, this allegation has neither safety significance nor generic implications.

   The TRT was unable to learn the identity of the alleger during the inspection; therefore, no followup interview with this alleger was possible.

6. Actions Required: None.

1. Allegation Category: Miscellaneous 2, Reactor Pressure Vessel

2. Allegation Numbers: AM-3 and AM-23b.

3. Characterization: Two allegations concerning the reactor pressure vessel (RPV) are characterized as follows:

   a. It is alleged that during hot functional testing (HFT) expansion caused the reactor pressure vessel reflective insulation (RPVRI) to come in contact with the concrete biological shield wall (AM-3).

   b. It is also alleged that the Unit 1 RPV is located 3/16 inch to the west of the north-south centerline through the containment building (AM-23b).

4. Assessment of Safety Significance: The NRC Technical Review Team (TRT) reviewed the Final Safety Analysis Report (FSAR) for verification of RPV reflective insulation quality class; background material in the NRC Inspection Report 50-445/83-34; 50-446/83-18 and the affidavit of Doyle Hunnicutt clarifying this report; Brown & Root (B&R) construction operation traveler 35-1195, and Westinghouse (W) Field Change Notices (FCNs) TBXM-10609, 10611, and 10612. The TRT also reviewed photographs of construction debris obstructions between the RPVRI and the biological shield and discussed all of these documents with the Texas Utilities Electric Company (TUEC) Mechanical Engineering Supervisor, the W project manager, and the TUEC heating, ventilation, and air conditioning system (HVAC) startup group leader. The TRT also reviewed documentation for hot functional test (HFT) No. ICP-PT-5502 for any information concerning the allegations, but found no specific notations about them.

   The Comanche Peak Steam Electric Station (CPSES) FSAR, Volume XIV, Appendix 17-A, classifies the RPVRI as nonsafety-related. The support ring (a support channel) for the RPVRI is part of the RPVRI; therefore, it too is considered to be nonsafety related. The only function of this support channel is to support the refective insulation. It is not needed for the safe shutdown of the plant but simply insulates the vessel. In this case it assumed importance only because of its effect on adjacent safety-related structures or components. A Westinghouse FCN (TBXM-10609) dated September 27, 1983, documented an unacceptable condition which was identified during hot functional testing. Actual air temperatures during HFT were 288°F near the reactor vessel flange versus 150°F maximum in the cooling annulus between the RPVRI and the shield wall. Similarly unacceptable temperatures were noted (150°F actual vs 135°F allowable) in the ex-core detector wells. TUEC letter TX5054 reported extreme temperatures up to 314°F. Further examination by W personnel revealed that cooling air flow was restricted by the RPVRI support channel because of construction debris between the RPVRI support channel and the steel-lined concrete biological shield wall and because of restriction by the support channel. The debris was removed by a remote technique and the gap was fiberoptically inspected. Instead of the nominal design gap (cold) distance of 7/8 inch between the support channel and the shield wall, this inspection identified cold air gaps as follows: a 7/8-inch air gap extending one quarter around (90°)

the annulus circumference, a 1/2 inch air gap extending one half (180°) around the circumference, and a 1/4-inch air gap extending around the remaining quarter (90°) of the circumference.

The TRT discussed the field change notice with the TUEC Project Mechanical Engineer and the Westinghouse Site Manager. Specifically, the staff asked the reason for there being too little space between the reactor vessel insulation support channel (which supports the reflective insulation) and the biological shield wall. The TRT learned that the original Westinghouse specification required the support channel to be inside the insulation, but TRANSCO Inc., the vendor, requested a design change to permit the support channel to be placed outside of the insulation. Gibbs & Hill Inc., the Architect Engineer, did not incorporate this change into the design nor did they consider the impact on the cooling of the reactor cavity; thus, there was too little clearance between the outer circumference of the support channel and the shield wall, which resulted in restricted air flow and overheating.

During the HFT, TUEC identified and recorded the inadequate cooling as a test deficiency. On July 29, 1983, they reported the HFT deficiency (orally) to the NRC and formally reported the deficiency as a 10 CFR Part 50.55(e) item on August 25, 1983, completing their reporting in TUEC letter TXX-4054, dated September 26, 1983. To correct this deficiency TUEC modified the insulation support channel by cutting holes in the top and bottom flanges of the channel to allow sufficient air flow and heat removal and to ensure proper operation of the ex-core detectors and protection of the biological shield wall. TUEC made this modification in accordance with W procedure MP 2.7.1/TBX-3, dated October 1, 1983. In addition, the existing insulation seals and heat removal capacity were improved. Further discussions with the TUEC engineer revealed that air flow tests had been performed since the Unit 1 support channel was modified; however, final results of these tests will not be known until the HFT is redone. The retest was scheduled, completed, and is being evaluated.

The TRT found that the 50.55(e) report of the corrective action taken regarding this deficiency did not include determination of the underlying cause of the deficiency. In addition, the report included no discussion of the effect on Unit 2 or how such a deficiency could be prevented in Unit 2. However, TUEC did fiberoptically inspect Unit 2 for debris or a similar gap, and found no problems. The TRT determined that the areas where debris entered the gap have been sealed in both Units 1 and 2, and TUEC anticipates no further problems with debris.

The TRT also evaluated the allegation concerning improper placement of the Unit 1 RPV by reviewing RPV construction operation traveler (COT) No. 35-1195 and the field survey note, "Final Setting Alignment," dated June 29, 1978. In addition, the TRT reviewed the W "Mechanical Service Manual," which includes procedures for setting the RPV, with the W project manager to determine the importance of the location of the RPV relative to the centerline of the Containment Building. The W manual states that alignment with other nuclear steam supply system (NSSS) components is the most critical location factor, and alignment with the center of the Containment Building is of secondary importance.

The Unit 1 RPV is alleged to be misaligned by 3/16-inch; however, operation No. 7 of COT No. 35-1195 shows a setting tolerance of ± 1/4 inch. The alleged 3/16-inch misalignment is within this tolerance. Attachment 35-1195-MCP-1 of COT No. 35-1195, "Alignment Location Record - Alignment Final Set," shows the maximum RPV deviation from the north-south and east-west centerlines of the Unit 1 Containment Building to be 0.003 inches which is within the specified tolerance. The TRT reviewed installation records and determined that the critical relationship of the NSSS components to each other, as well as to the Containment Building centerlines (N-S, E-W), was accurately maintained during installation of the reactor pressure vessel.

5. <u>Conclusion and Staff Positions</u>: Based on review of documentation and discussions, the TRT concludes that the RPVRI did make contact with construction debris, but did not contact the steel-lined concrete biological shield wall as specifically alleged. During fiberoptic inspection, TUEC personnel observed no visible damage to the reflective insulation, and all corrective modifications were accomplished and accepted in accordance with procedure MP 2.7.1-TBX-3 and FCNs TBXM-10609, 10611 and 10612.

   The allegation, as specifically stated, cannot be substantiated, although it does have some merit because an unsatisfactory condition did exist in that the reflective insulation made contact with debris. However, this allegation has both safety significance and generic implications because of peripheral issues; i.e., failure to assure that proper design changes were communicated between organizations, failure to determine and report the underlying cause of a significant deficiency, and failure to ensure a proper gap between the support channel and shield wall when the vessel was set.

   The TRT also concludes that the RPV is set within the design location tolerance. Therefore, this allegation is not substantiated and has neither safety significance nor generic implications.

   The TRT is unable to interview the alleger to provide its findings and conclusions because the identity of the alleger is unknown. The TRT could not identify the alleger responsible for allegation AM-3 because an anonymous alleger called the Dallas <u>Times-Herald</u>. The identity of the alleger of AM-23 is unknown because the person who received the allegation did not record the alleger's name and no longer remembers it.

6. <u>Actions Required</u>: TUEC shall:

   1. Review their procedures for approval of design changes to nonnuclear safety-related equipment, such as the RPVRI, and make revisions as necessary to ensure that such design changes do not adversely affect safety-related systems.

2. Review procedures for reporting significant design/construction deficiencies, pursuant to 10 CFR Part 50.55(e), and make changes as necessary to ensure that complete evaluations are specified.

3. Provide analysis which verifies that the cooling flow in the annulus between the RPVRI and the shield wall of Unit 2 is adequate for the as-built condition.

4. Verify during Unit 1 hot functional testing that completed modifications to the RPVRI support ring now allow adequate cooling air flow.*

The TRT notes that control of debris in critical spacings between components and/or structures was identified as an issue both in the investigation of this allegation and in the civil and structural area (Item II.c, "Maintenance of Air Gap Between Concrete Structure"), contained in Darrell G. Eisenhut's September 18, 1984, letter to TUEC (Attachment 3). Accordingly, TUEC shall also:

1. Identify areas in the plant with spacing between components and/or structures that are necessary for proper functioning of safety-related components, systems, or structures in which unwanted debris may collect and be undetected or be difficult to remove.

2. Inspect the areas and spaces identified and remove debris.

3. Institute a program to minimize the collection of debris in critical spaces and periodically reinspect these spaces and remove any debris which may be present.

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*The test has been completed. However, TUEC's analysis of test results is still underway.

1. Allegation Category: Miscellaneous 3, PSAR Errors

2. Allegation Number: AM-4

3. Characterization: It is alleged that Sections 10.2-11 and 10.2-12 of Volume VII of the Comanche Peak Preliminary Safety Analysis Report (PSAR) contain errors.

4. Assessment of Safety Significance: The NRC Technical Review Team (TRT) could not contact the alleger because the alleger's telephone number and address were unknown and relatives would not provide the TRT with this information.

   The TRT determined that this allegation refers specifically to PSAR Section 10.2-11 and 10.2-12; however, in the alleger's statement to the NRC Region IV Office of Investigation, the terminology "FSAR" (Final Safety Analysis Report) was frequently used when the term "PSAR" apparently was intended. The alleged errors refer to turbine missile energy calculations in PSAR Volume VII, Sections 10.2-11 and 10.2-12. Sections 10.2-11 and 10.2-12 of the FSAR do not pertain to turbine missile energy.

   The referenced PSAR sections contain a bounding design analysis calculation used to determine the maximum energy available if a turbine failure occurs. Using the equation, $E_{mf} = \frac{1}{2}\,mv^2 = \frac{1}{2}\,w/g_c\ v^2$, the calculated value should be 18.27×10^6 ft-lbs; however, the PSAR erroneously stated the calculated value as 18.27×10 ft-lbs. It appears that the exponent was dropped because of a typographical error.

   The PSAR reflects the preliminary plant design, but it is the FSAR that reflects the final plant design and safety analysis. Sections 10.2, 3.5.1.3, and 1.3 (Table 1.3-2, page 27) of the FSAR contain the final design and analysis for the turbine generator. These sections do not contain the alleged erroneous calculation. The FSAR refers to "Allis-Chalmers P.S. Inc., ER-503, 'Turbine Missile Analysis for 1800 t/min NSTG with 44-inch Last Stage Blades,' July 1985." The analysis in this document is specifically applicable to the turbine generator which was actually purchased. The TRT reviewed the Allis-Chalmers analysis in the FSAR and determined that it corrected the turbine missile energy calculations in the PSAR.

   The TRT found no errors in the Allis-Chalmers analysis of turbine missile probability, and concurred with the Texas Utilities Electric Company (TUEC) conclusion that even if missiles were generated, they would be contained by the low pressure turbine casing.

   Although the alleged error appeared to be a typographical error, the TRT randomly selected calculations from other FSAR sections and paragraphs (3.5A-1, 3.5-16A, and 10.3-5) for review and found no additional errors.

5. Conclusion and Staff Positions: The TRT reviewed the PSAR and determined that there was an error in the PSAR as alleged, but that it appeared to be a typographical error. However, a review of the FSAR showed that the alleged error in the preliminary design calculation of the PSAR had been corrected in the FSAR. The NRC staff also randomly reviewed other calculations in the FSAR and found that they were free of error. Additionally,

the FSAR has been reviewed in detail by the NRC Office of Nuclear Reactor Regulation (NRR) as part of the licensing process and this included review of the turbine missile analysis. Accordingly, this allegation has neither safety significance nor generic implications.

The TRT contacted the alleger to discuss its findings and conclusions; however, the alleger declined to participate in the planned followup interview. A letter was sent to the alleger on January 22, 1985, requesting that he reconsider participating in an interview with the TRT; however, there has been no response to this letter.

6. Actions Required: None.

1. Allegation Category: Miscellaneous 4, Radioactive Material Release

2. Allegation Number: AM-5

3. Characterization: It is alleged that someone "threw something radioactive in the lake," that is, in the Comanche Peak Reservoir, sometime between September and November 1978, and that this material may have been tritium.

4. Assessment of Safety Significance: The NRC Technical Review Team (TRT) reviewed the background material contained in NRC Region IV Inspection Report No. 50-445/79-22. 50-446/79-21 which documents an inspection conducted September 7 and 12 and October 11, 16, 18 and 20, 1979, which followed up on allegations that appeared in the University of Texas at Arlington (UTA) newspaper, the "Shorthorn," on Wednesday, July 18, 1979. This inspection concluded that "this allegation appears to have no merit." The TRT also reviewed an NRC Office of Investigation transcript of interview No. 84-006, dated March 7, 1984. From this document, the TRT learned that the alleger stated that he was looking through his turbine book and he saw that they used tritium gas to inspect for leaks. The alleger connected this with a story that another worker had told him, i.e., that one night the other worker was working down there when they threw something radioactive in the lake. The alleger in turn contacted the NRC to provide this information.

   The TRT interviewed both the TUEC Radiation Protection Engineer (RPE) and the Radiation Protection Supervisor (RPS) about the allegation and about receipt of radioactive material at the Comanche Peak Steam Electric Station (CPSES), and also reviewed their radioactive source log. The RPS stated, and the radioactive source log documented, that first receipt of radioactive material at CPSES was on January 10, 1980, and that the quantity of strontium 90 and tritium received was exempt from licensing requirements. These small quantities were for laboratory use. The RPS also stated that although a leak-test procedure utilizing tritium could be employed for the turbine-generator unit, the turbine generator at CPSES was hydrotested during hot functional testing, in lieu of the tritium leak-test procedure outlined in the manufacturer's instruction manual.

   The RPS stated that the first controlled shipment of tritium at CPSES was received on January 31, 1983. This shipment was authorized under a state of Texas radioactive material license (No. 5-2892) issued October 9, 1980, and the CPSES radioactive source log documented this receipt. The RPS also stated that an aliquot of this tritium standard was used to prepare a standard to calibrate the tritium monitors located on the turbine-generator unit. The aliquot of tritium used to prepare the calibration standard was recovered, and CPSES verified its original radioactivity. The RPS stated that when the plant returns to hot functional testing after fuel load, the primary coolant of the turbine generator will be "spiked" with this tritium. During operation of the turbine generator, a small amount of hydrogen gas will be extracted and measured by a tritium monitor. Thus, if there was a leak within the turbine generator cooling system, tritium would be detected in the hydrogen gas.

The TRT reviewed a Texas Utilities Generating Company (TUGCO) startup test log, which documented that the turbine generator primary coolant system and components were pressure-tested, not tritium-tested, in accordance with test procedure CPM 6.9I, "Main Generator Primary Water and Seal Oil." These tests began on December 4, 1982, and concluded on October 11, 1983.

The TRT reviewed the CPSES-established documented program for controlling radioactive source material, as outlined in health physics administrative procedure (HPA)-105; for receipt of radioactive material, as outlined in health physics instruction (HPI)-202; and, for shipment of radioactive materials as outlined in HPI-203. The TRT also learned that the Region IV staff conducted extensive reviews of these programs as part of the pre-operation inspection program, and documented the results of these inspections in Region IV Inspection Reports 50-445/83-16, 83-35, 84-02, and 84-25. The TRT found no evidence to support the mishandling allegation.

5. <u>Conclusion and Staff Positions</u>: Based on reviews of CPSES procedures, radioactive source log sheets, startup test logs and startup test data sheets, and on interviews with the RPE and the RPS, the TRT concludes that CPSES received radioactive material (which was exempt from NRC licensing requirements) approximately in January 1980, but did not receive a licensed shipment of tritium prior to January 1983. The NRC staff found no evidence to support the allegation that radioactive material was "dumped" into the Comanche Peak reservoir at CPSES during the period from September 1978 to November 1978.

   The TRT tried to contact the alleger during the inspection; however, the alleger was unavailable. Therefore, the TRT was unable to provide the above findings and conclusion to the alleger. A letter was sent to the alleger on January 22, 1985, requesting that he participate in an interview with the TRT; however, there has been no response to this letter.

6. <u>Actions Required</u>: None.

1. Allegation Category: Miscellaneous 5, High Pressure Turbine

2. Allegation Number: AM-6

3. Characterization: It is alleged that cracks were observed in the lower casing of the high pressure (HP) turbine. The alleger did not specify the reactor unit where the cracks were alleged to exist.

4. Assessment of Safety Significance: The NRC Technical Review Team (TRT) could not contact the alleger because the alleger's telephone number and address were unknown and relatives would not provide the TRT with this information.

   The TRT reviewed the applicable design, procurement, and vendor inspection information described in the Final Safety Analysis Report (FSAR), Section 10.2, "Turbine Generator"; Allis-Chalmers, Turbine Description, No. 1-1-0200-7163; TUGCO Purchase Order, No. CP-0003; and, Vendor Surveillance Report No. 294. The HP turbine was supplied by Allis-Chalmers Power Systems, Inc., and is classified as nonsafety related. The HP casings were fabricated in Japan by a casting process; however, they were subsequently shipped to the Mulheim plant in Germany where manufacturing was completed.

   The vendor surveillance referenced above was performed in Germany by a Gibbs & Hill (G&H) QC engineer on October 3-6, 1977. Records indicate that the engineer visually examined the Unit 1 HP turbine, witnessed hydrostatic testing, and reviewed the supporting documentation, which included nondestructive examination (NDE) records. The vendor surveillance report on the Unit 1 HP turbine concluded that inspection and testing showed the casing to be satisfactory.

   On August 4, 1984, the TRT interviewed both the Texas Utilities Electric Company (TUEC) supervisor of turbine construction and the TUEC lead startup engineer. They stated that they observed no cracks in the casing. In addition, the lead startup engineer was present when the Unit 1 turbine was rolled to synchronous speed during testing, and he indicated that no casing problems (including casing leaks) were observed and that only an insignificant diaphram leak was detected during testing. The TRT also reviewed test documentation which showed acceptable results.

   The TRT inspected the outside of the upper and lower casing of the HP turbine for Unit 2 and found no cracks. They did not inspect the casing of the Unit 1 HP turbine because it is nonsafety related and is wrapped with insulation. Since independent vendor inspection and test records, and TUEC observation and testing, revealed no unacceptable conditions, the TRT did not request removal of the insulation or did not perform further casing inspections of the Unit 1 turbine.

5. Conclusion and Staff Positions: The TRT determined that TUEC and vendor personnel performed visual inspections, witnessed hydrostatic and startup testing on the HP turbine for Unit 1, and found no unacceptable conditions. The TRT also inspected the outer casing of the Unit 2 turbine and found no cracks. Moreover, this equipment is classified as nonsafety related and is not needed for the safe shutdown of the reactor.

The TRT attempted to contact the alleger during the TRT inspection to obtain specifics, but the alleger declined to be interviewed. Consequently, the TRT was unable to present its findings and conclusions to the alleger in a follow-up interview. A letter was sent to the alleger on January 22, 1985, requesting that he recondiser participating in an interview, with the TRT; however, there has been no response to this letter.

6. Actions Required: None.

1. Allegation Category: Miscellaneous No. 6, Pressurizer Area Piping

2. Allegation Number: AM-7

3. Characterization: It is alleged that an 18-inch section was cut from a prefabricated pipe "in the pressurizer area."

4. Assessment of Safety Significance: The NRC Technical Review Team (TRT) attempted to contact the alleger to determine the exact location of the cut piping because it was not specifically identified in the allegation. However, the alleger's telephone number and address were unknown, and relatives of the alleger would not provide the NRC staff with this information.

   Due to the size of the piping (18 inches), and because the alleger stated that the "cut pipe was close to the pressurizer," the staff assumed that the cut section of piping was located in the pressurizer system. Based on a review of a Final Safety Analysis Report (FSAR) flow diagram ("Reactor Coolant System," Section 5.1) and a walkdown inspection of the pressurizer area, the NRC staff determined that there is no 18-inch line in the pressurizer system and immediate area and that the 12-inch piping that runs from near the top of the pressurizer (at approximately the 907-foot elevation) to the pressurizer relief tank (at the 834-foot elevation) most closely fit the description given in the allegation.

   The TRT also reviewed the isometric drawings for Unit 1 and Unit 2. These show the as-built configuration of the four 6-inch lines which run from the pressurizer to a common 12-inch line which, in turn, runs to the pressurizer relief tank. In this run of piping are both smaller piping and a 3-inch line which runs from the 12-inch line to the safety relief valves, and which ties into piping in the residual heat removal system (RHR) suction (Trains A and B) and for chemical and volume control system (CVCS) seal return and letdown. All modifications to the pipe lines identified in the isometric drawings, including the trimming of pipe for fitup, were recorded on component modification cards (CMCs). Trimming modifications of 7/8-inch and 1-1/2-inch made on 12-inch piping were recorded on CMCs 61551 and 47943R1. The TRT inspected and counted the number of welds in all of the above piping runs, then compared that number with both the number shown on the as-built isometric drawings and the number on the CMCs. All three numbers were the same, and the NRC staff found no evidence of unauthorized work in the piping system.

   Because the 12-inch piping runs are nonsafety related and not essential for safe shutdown of the plant, the quality assurance requirements of 10 CFR Part 50 Appendix B are not applicable; however, Texas Utilities Electric Company (TUEC) technicians monitored the piping installation in accordance with CP-CPM-6.9, "Welding and Related Processes." The TRT selected and reviewed two welding records and two test reports which confirmed that appropriate procedures were used to control welding. Again, the TRT found no evidence of unauthorized work.

TUEC performed a successful final system check from the pressurizer to the pressurizer safety relief valves, including the piping down through the RHR suction and the CVCS seal return letdown relief valves. The system check was made concurrently with hydrostatic tests IRC-101 and IRC-01A on August 31 and October 19, 1982. The actual test pressure was 1113 psig for 21 minutes, which meets the ANSI B31-1 code requirement which is 1.5 times the design pressure of 700 psig or 1050 psig.

The diameter of the Class 1 ASME pressurizer piping did not fit the description contained in the allegation. However, the TRT inspected the Class 1 piping from the pressurizer to the upstream side of the pressurizer safety relief valves and reviewed the as-built isometric drawings, nonconformance reports (NCRs), CMCs, and N-5 ASME data forms for the 3-inch and 6-inch piping to ensure that modifications or changes had been authorized and were recorded. Based on this review, the TRT found no evidence of unauthorized work and determined that the QA records for ASME piping were in order. In addition, the NRC staff inspected the pipelines visible at floor level, and the number of welds appeared to correlate with the as-built isometric drawings and with the information on the N-5 data forms required by ASME Code.

5. Conclusion and Staff Positions: Based on inspections of the piping and on a review of applicable documents, the TRT found no evidence which would support the allegation that unauthorized cuts or welds were made in piping from the pressurizer to the pressurizer relief tank, the RHR suction relief valves in Trains A and B, the CVCS seal return, or the CVCS letdown piping systems close to or in the pressurizer area. Accordingly, this allegation has neither safety significance nor generic implications.

   The TRT was unable to provide its findings and conclusions to the alleger, because the alleger declined to be interviewed. A letter was sent to the alleger on January 22, 1985, requesting that he recondiser participating in an interview, with the TRT; however, there has been no response to this letter.

6. Actions Required: None.

1. Allegation Category: Miscellaneous 7, Condenser

2. Allegation Number: AM-8, AM-9 and AM-10

3. Characterization: It is alleged that: (a) the Unit 1 main condenser tubes were beaten with air and sledge hammers, were split during belling and flaring, and were improperly rolled; (b) the wrong type condenser for the steam generator was used; (c) the tube support sheets had holes that were misaligned by 3/8 inch; and, (d) the turbine-to-condenser tubing was misaligned and jacked into alignment, causing stress.

4. Assessment of Safety Significance: The NRC Technical Review Team (TRT) interviewed the alleger on August 24, 1984, to gather additional information regarding his concerns.

   A Region IV letter to TUEC identified the potential safety concerns associated with this allegation and asked Texas Utilities Electric Company (TUEC) to respond in writing. In a letter dated July 9, 1984, TUEC outlined their monitoring procedures for condensers during and after construction activities, as well as the various tests that they had conducted. This letter stated that testing demonstrated that no leakage of lake water into the steam generator occurred because of differential pressure that is maintained during operation. It also states that the main condenser and circulating water systems are not safety related and therefore not subject to the quality assurance requirement of 10 CFR Part 50, Appendix B. Regardless, construction and fabrication were done in accordance with sound engineering and construction practices.

   In assessing this allegation, the TRT reviewed NRC Office of Investigations interview 84-006; the RIV letter and TUEC response referenced above; TUEC office memoranda SU-81051, SU-81081, and SU-81134; and Condenser Retest No. 1 CP-AT-27-01. These documents show that TUEC was well aware of and was controlling and correcting problems associated with the condenser fabrication and operation.

   The TRT attempted to visually inspect the condenser internals, but this was not possible because of the congestion caused by the tube bundles, the internal piping, and the bracing. Therefore, the inspection was limited to using a hand-held light and observing the bundles from the manway opening. Observation was difficult because of the distance between the manway and the tube bundles and because of surface rust; however, the limited inspection detected no irregularities.

   The TRT interviewed three employees who had direct knowledge of the work on the main condenser tubes. With respect to part (a) of this allegation, i.e., use of hammers and splitting tubes, a Brown & Root (B&R) Millwright Superintendent stated that hammers were used on the condenser tubes, but only after the installation of a special tool to protect the tube ends. He also stated that this is a standard practice in condenser assembly. Concerning part (c) of this allegation, the Millwright Superintendent disclaimed any knowledge of misaligned tube sheet holes.

   The TRT also interviewed a TUEC Operations Results Engineering Supervisor concerning the availability of documents such as inspection reports, deficiency reports, in-progress monitoring reports, nondestructive examination (NDE) reports, and procedures related to the allegation. He stated that work was performed to B&R procedures and that he did not keep any official records because there was no requirement; therefore his particular involvement was limited to nonsafety-related surveillance for commercial consideration. When asked about tube-rolling problems, he stated that the variance in manufacturing tolerances of the tube outside diameters and the tube sheet hole diameters did present a problem, part (c), in the beginning, but the problem, part (a) of this allegation, was solved when they properly calibrated the rolling tool. He also stated that as the work progressed, his level of confidence in the craft personnel's work reached a point where 100 percent surveillance was not required. He added that during the rolling of the tubes, the optimum tube-wall reduction for a positive tube-to-sheet seal was in the range of 6 to 9 percent and that in the beginning, when experimenting with the torque on the rolling tool, although the 9 percent reduction in wall thickness did occur in some cases, it did not result in any of the tubes cracking.

In another interview, a B&R Millwright Foreman informed the TRT that he was not involved in the construction of the condenser and disclaimed any knowledge of the alleged problems. Part (b) of this allegation was discussed with the Foreman, who said it was rumored that the condensers, the auxiliary condensers, and the moisture separators might be retubed. This rumor was confirmed both by the Millwright Superintendent and in a telephone conversation with the TUEC Nuclear Engineering Manager. The latter informed the TRT that his group was commissioned to make a feasibility study of condenser retubing, first doing Unit 2 and then doing Unit 1 during the first refueling outage. The proposed change was to retube using titanium tubes instead of chrome-nickle (Cr-Ni) tubes in order to raise the pH level of the water on the secondary side from a level of 9.4 to between 9.8 and 10.0. The alleger may have been referring to this proposed change when he said "it's the wrong type condenser for the type of steam generator." However, Westinghouse provided Gibbs & Hill, Inc. (G&H) with Specification 2323-MS-23, which contained the original design, operating conditions and criteria that were to be met, and this specification was used. The second design of the tubing was proposed as a design improvement.

On September 14, 1984, the TRT contacted an Allis-Chalmers (A-C) representative who supplied information regarding part (d) of this allegation. Installation progress reports showed that the low-pressure (LP) II condenser's expansion-joint weld was completed on September 18, 1978. The welding process was monitored during welding, and once in approximately every hour micrometer readings were taken at 12 points around the joint. These readings showed that the maximum movement was 0.026 inches (0.660 mm), which was acceptable to Allis Chalmers, Westinghouse, and TUEC. Other than visual inspection, an NDE was not required.

The same progress report showed that the LP-I expansion joint welding began on the same date. On October 13, 1978, deviation report No. 14094 was written against the lower weld on this joint. The movement of 0.277 inches (7.030 mm) exceeded that allowed by A-C. Consequently, on October 19, 1978, the removal of the lower weld was begun, and some 1,500 inches of 3/4-inch weld were removed. The micrometer logs kept by the TUEC Operations Maintenance Foreman showed that rewelding began on November 13, 1978, and was completed on November 17, 1978, and that movement was controlled to a maximum of 0.023 inches (0.584 mm). All welding was done to B&R weld procedure (No. 10046) and to Westinghouse directions for skip (intermittent) welding. Considering these welding controls, there was no evidence that jacking occurred or that any undue or undesirable stress was introduced into the welded joints. Following welding, the only NDEs performed (other than visual inspections) were hydrostatic and vacuum tests which were successfully completed.

The TRT also reviewed seven condenser hydrostatic test packages for the results of tests conducted in accordance with the G&H test procedure (Specification 2323-M-23). These tests spanned the period from December 1980 to April 1984. The TRT found that retests of the shell were made following design changes. The last of these retests, No. ICO-0200E, was performed in September 1983, and involved sodium tracer injection and sampling at points through the condenser wall. In all cases the tests were successfully completed. In May 1984, TUEC performed a vacuum and water box priming retest (Procedure ICP-AT-27-01-RT-1) to again verify that the main condenser could be evacuated by the vacuum pumps and hold vacuum for 1 hour. The system test engineer witnessed and accepted this sucessfully completed test. The lead startup engineer, the manager of plant operations, the TUSI nuclear engineering manager, and the manager of nuclear operations also reviewed and accepted the test results. The TRT's review of these documents indicated that there was no evidence of an overstress problem with either the expansion joint or the piping connection.

5. <u>Conclusion and Staff Positions</u>: The TRT determined that this unit is classified as nonsafety related and is not essential for the safe shutdown of the plant, and confirmed that when TUEC began the tube-rolling procedure, they experienced some fabrication problems; however, these problems appear to have been solved. The TRT also confirmed that TUEC plans to retube the condenser with titanium tubes to improve its design and operation.

The TRT found no evidence that holes in the tube support sheets were misaligned by 3/8-inch, nor did they find evidence that the turbine to condenser was misaligned to the point that excessive stress was introduced. The conclusive evidence is that the condenser was constructed following approved construction and testing procedures and, as such, will perform its design function.

The TRT concludes that this allegation was not substantiated. However, it was true that fabrication problems occurred and that condenser redesign (tube material changes) and misalignment occurred, but not as alleged. Accordingly, this allegation has neither safety significance nor generic implications.

The TRT attempted to present its finding and conclusions to the alleger in a follow-up interview, but the alleger could not be located. The TRT was unable to find either a telephone number of an address for the alleger. A letter will be sent to the intervenor of record, outlining the resolution of the alleger's concerns.

6. <u>Actions Required</u>: None.

1. Allegation Category: Miscellaneous 8, Damaged Component Cooling Water Tank Supports

2. Allegation Number: AM-12

3. Characterization: It is alleged that during the installation of Unit 1 component cooling water (CCW) surge tank, the anchor bolts were damaged.

4. Assessment of Safety Significance: The NRC Technical Review Team interviewed the alleger on August 24, 1984, to gather additional information regarding his concerns.

   The TRT reviewed the background material which alleged that the anchor bolts were beaten sideways with a hammer to make them line up with holes in the plate and were overtorqued to the point that they stretched. The CCW surge tank, part of the component cooling water system, is listed in Final Safety Analysis Report (FSAR) Table 17A-1 as Safety Class 3, ASME III Code Class 3, and Seismic Category I. The centerline of the CCW surge tank for Unit 1 is at an elevation of 889 feet, 6 inches in the Auxiliary Building; the baseplate attaches to bolts embedded in concrete which interface at an elevation of 895 feet, 6 inches.

   In assessing this allegation, the TRT reviewed Texas Utilities Services Inc. (TUSI) Drawing N-2640-359. The TRT also visually inspected the installed CCW surge tank and saw no stripped threads or bent or cracked bolts. Two nuts and one washer were present on each bolt.

   The CCW surge tank has 10 bolts on each end which support the tank. A review of the installation documentation revealed that 5 out of the 20 concrete anchor bolts were misaligned. A TUSI letter (CPP-00825), dated March 2, 1979, documents the need for modifying the baseplate holes. This letter, which describes the misalignment, indicates that the bolts were not installed as required by the specification and drawing. As a result, component modification card (CMC) No. 4263 was approved June 8, 1979. The TRT found no nonconformance reports (NCRs) in the quality assurance (QA) records vault. The misalignment may have caused installation problems; however, if any damage occurred at that time it was not documented.

   Traveler No. ME78-108-1101 and a traveler revision record sheet document the installation of the CCW surge tank. This traveler was initiated on October 10, 1978, and the tank was placed on June 13, 1979, when the Millwright Supervisor signed the traveler. This traveler failed to give instructions for tightening nuts on anchor bolts as required by Procedure 35-1195-MCP-1, Revision 2, paragraphs 4.1.10 and 4.1.11. Both the quality control (QC) inspector and the Millwright Supervisor signed the traveler on June 18, 1979, indicating that the tank was level and located as recorded on the as-built drawing.

   The TRT interviewed the Millwright Supervisor and the QA/QC and engineering personnel listed on the traveler and learned that although they had direct knowledge of work activities on a day-to-day basis, they had no knowledge of bent or cracked bolts or of damaged threads. The engineer stated that the millwrights routinely brought any problems to him, and he could not believe craftsmen assigned to him would carelessly bend and damage the bolts.

When reviewing this allegation, the TRT found that design change authorization (DCA) No. 9909, Revision 1, dated April 10, 1981, documented modifications made to the structural support saddles to increase their strength and to meet seismic requirements. Travelers No. CE-82-143-1100 and No. ME-81-1563-2-1101 documented that this work was done on Units 1 and 2. DCA No. 11468, Revision 10, dated May 1, 1984, documented additional seismic brace work on Units 1 and 2 tank supports. In their review of the applicable documentation, the TRT found no evidence indicating problems with the installation of the Unit 2 CCW surge tank or with misaligned or damaged bolts.

The TRT discussed the load on the anchor bolts used to install each of the Unit 1 and 2 tanks with the cognizant Gibbs & Hill Inc. (G&H) engineer, who provided and explained the data in G&H Calculation Number SAB-104, Set 4. The full tank load was calculated to be 51.5 ksi (1000 lb/square inch). Based on these values, the worst case analysis for a seismic event was calculated to determine the tensile and shear loads on the 1-inch anchor bolts (ASME A320-A7, 105,000 ksi yield strength). These bolts and the bracing and supports, added as a result of modifications, were documented as being strong enough to carry the loads and meet safe shutdown earthquake requirements. Nuts used to bolt the west end of the tank to the concrete beam are required only to be hand-tight with a locknut; therefore, no torquing was required. On the east end, torquing and a locknut are required.

5. Conclusion and Staff Positions: Based on its review of applicable documentation and interviews with cognizant personnel, the TRT concludes that problems were experienced during installation of the Unit 1 CCW surge tank because of misaligned bolts and that the necessary modifications were made after engineering review and approval. The TRT found no evidence to support the allegation that these bolts were beaten with hammers and were overtorqued to the point of stretching and cracking them. Accordingly, this allegation has neither safety significance nor generic implications.

   The TRT tried to provide the above findings and conclusions to the alleger in a follow-up interview, but the alleger could not be located. The TRT was unable to find either a telephone number or an address for the alleger. A letter will be sent to the intervenor of record, outlining the resolution of the alleger's concerns.

6. Actions Required: None. However, if a violation is issued TUEC will be required to take corrective action and respond.

1. Allegation Category: Miscellaneous 9, Hayward Tyler Pump Deficiencies

2. Allegation Number: AM-13

3. Characterization: It is alleged that Comanche Peak Steam Electric Station (CPSES) has pumps in safety systems manufactured by Hayward Tyler Pump Co. (HTPC) that may have unidentified deficiencies because of a poor quality assurance (QA) program by HTPC.

4. Assessment of Safety Significance: The NRC Technical Review Team (TRT) interviewed the alleger on August 24, 1984, to gather additional information regarding his concerns.

   The NRC Technical Review Team (TRT) learned that in late 1981, the NRC received allegations that upper management of Hayward Tyler Pump Company of Burlington, Vermont failed to support the quality assurance program.

   In 1982, the NRC Region IV staff inspected the HTPC QA program after receiving these allegations. The investigation established that significant deficiencies existed in the implementation of HTPC's QA program from 1977 through 1981. As a result, the NRC issued a report and a Notice of Nonconformance on December 22, 1982. As a result of the above findings, the NRC also issued IE Bulletin No. 83-05 to licensees and applicants on May 13, 1983. This bulletin addressed HTPC's failure to effectively implement its QA program, and required that specific actions be taken by the holders of operating licenses and construction permits who were using or planning to use HTPC pumps in safety systems.

   IE Bulletin 83-05 requirements for successful pump operability applied to both the original pump assembly and to reassembled pumps which use spare parts. The bulletin required the following actions from NRC applicants and licensees:

   - to provide NRC with the number of HTPC pumps and their service application,

   - to provide NRC with a summary of the in-service test requirements for the affected pumps and spare parts,

   - to conduct a pump performance test by running the pump continuously for a minimum of 48 hours without maintenance or repair, with the test incorporating specific criteria provided by HTPC,

   - to provide NRC with the results of the required ASME Code hydrostatic pressure test,

   - to implement HTPC recommendations with respect to fitup and dimensional considerations during installation of spare parts,

   - to conduct a pump performance test when spare parts are installed, unless it could be demonstrated that the spare part in question would not affect any parameters that are measured, and to provide NRC with the results of the required ASME Code hydrostatic pressure test on spare parts that form part of the ASME Code pressure boundary.

On May 24, 1982, an NRC Region IV inspector testified before the Atomic Safety and Licensing Board (ASLB). On page 99 (Answer 64) the RIV inspector stated that Hayward Tyler pumps were used at CPSES and allegations or problems were being investigated. This was preceded by an allegation made by an unknown alleger in March 1982, and may have led to the ASLB questions about these pumps.

The TRT determined that an NRC inspector had reviewed the Texas Utilities Electric Company (TUEC) letter dated August 10, 1983, which was in response to IEB 83-05, including the supporting documentation, to assure that all required actions were addressed and had been performed. The inspector had determined that the use of HTPC pumps at CPSES is limited to the station service water systems and all required actions had been documented as complete with respect to the two CPSES Unit 1 station service water pumps (SSWP). The 48-hour endurance test requirements were met and exceeded when the Unit 1 SSWPs were operated continuously between March 3 and May 26, 1983. The number of hours accumulated during that run totaled 1528.75. The NRC inspector verified this by a review of the CPSES Unit 1 Control Room reactor operator log. TUEC's supervisor of technical support and startup engineering stated that the SSWPs had accumulated approximately 16,000 operational hours since initial startup in February/March 1981, without any major repairs or any significant degradation in performance.

The two Unit 2 SSWPs will be tested during the Unit 2 preoperational testing program, which incorporates the requirements of IEB 83-05.

5. Conclusions and Staff Position: The TRT concludes that TUEC had identified Hayward Tyler pumps onsite and tested the pumps and reported as required by IEB 83-05. The TRT also concludes that the allegation had potential safety significance and generic implications; however, TUGCO's compliance with IEB 83-05 has eliminated those concerns with respect to CPSES Unit 1 SSWPs. The Unit 2 pumps will be inspected under Unit 2 preoperational testing.

   The TRT attempted to provide the findings and conclusions described above to the alleger, but the individual could not be located. The TRT was unable to find either the telephone number or an address for the alleger. A letter will be sent to the intervenor of record, outlining the resolution of the alleger's concerns.

6. Actions Required: TUEC shall verify compliance with IEB 83-05 requirements for CPSES Unit 2 SSWPs during preoperational testing for Unit 2.

1. Allegation Category: Miscellaneous 10, Damaged Diesel Generators

2. Allegation Number: AM-14

3. Characterization: It is alleged that the Unit 1, Train A, diesel generator was damaged in May 1982, because of improper handling practices.

4. Assessment of Safety Significance: The NRC Technical Review Team (TRT) tried to contact the alleger during its inspection to learn more details about the allegation. The TRT reviewed all nonconformance reports (NCRs) issued in May 1982, that pertained to the emergency diesel generators (EDGs). Of the 11 NCRs reviewed, 4 (E-82-005335, E-82-00560, E-82-006065, and E-82-004795) documented equipment or instrument damage; however, the necessary corrective action was taken and the NCRs were closed appropriately.

   The TRT determined that it was unlikely that any damage that occurred in May 1982, could now affect the operability of the EDGs in light of the extensive EDG inspection and testing that took place in 1984. This inspection and testing was the result of generic EDG problems which Transamerica Delaval, Inc. (TDI) and owners of TDI's EDGs identified to the NRC.

   On August 12, 1983, the main crankshaft on one of the three EDGs at Shoreham Nuclear Power Station broke into two pieces during a load test. TDI issued several 10 CFR Part 21 reports that reflected a variety of major and minor defects. These defects included cracks in piston skirts, push rod cracks, governor drive coupling failures, potential failures in fuel lines, and dimensional problems with component fasteners and dowel pins. Although there are some design differences between the EDGs at Comanche Peak Steam Electric Station (CPSES) and those at other plants, the identified defects were generic in nature.

   During the evaluation of the failure and repairs of the Shoreham EDGs, information related to the operating history of TDI engines and the QA program of the manufacturer was identified which called into question the reliability of all TDI diesels. As a result of the evaluation and its generic implications, representatives from affected nuclear power plants formed an "Owners' Group" to investigate all aspects of the quality and reliability of TDI-supplied EDGs.

   The Owners' Group developed a generic inspection program. This program addressed the specific concerns brought about by defects reported to the NRC by owners of TDI EDGs, 10 CFR Part 21 reports from TDI, and other areas of concern in order to develop adequate confidence in these EDGs.

   Texas Utilities Electric Company (TUEC) also expanded some inspections of the TDI EDGs by increasing sample sizes, inspecting other areas on their own initiative, and inspecting both of the Unit 1 EDGs between February and June 1984. The inspections included disassembly and nondestructive examinations of parts using methods such as radiography, liquid penetrant testing, magnetic particle testing, visual inspections and measurements, eddy current testing, ultrasonic testing, and metal comparator testing. TUEC then transmitted the results of their inspections to the Owners' Group for evaluation and incorporation into the recertification process.

The EDGs were reassembled after cleaning, and the inspections and nondestructive tests were completed. This effort was closely controlled by approved procedures and by QC surveillance. TUEC replaced parts which had been identified as containing potential generic defects and also replaced those parts found to have defects previously unidentified.

Upon completion of assembly, TUEC retested each EDG by performing the entire portion of the preoperational test program which involved operation of the EDGs. The TRT determined that TUEC made the following tests and that the test results were satisfactory.

1CP-PT-29-01, RF1	"Diesel Generator Auxiliary Systems, Retest 1"
1CP-PT-29-02	"Diesel Generator Control Circuit Functional and Start Test"
1CP-PT-29-03	"Diesel Generator Load Tests"
1CP-PT-29-04	"Diesel Generator Sequencing and Operational Stability Test"
1CP-PT-29-05	"Diesel Generator Reliability Tests"

The TRT learned that the NRC Region IV (RIV) Resident Inspector for Operations conducted inspections on nearly a daily basis, starting with the disassembly process in February 1984, and ending with the witnessing of the testing in August 1984. This NRC inspection effort included (but was not limited to) observation of the work and testing in progress, review of procedures used and compliance thereto, and tracking the work to ensure that TUEC followed the Owners' Group program and adequately documented results. NRC Inspection Reports 50-445/84-07, -15, -17, -18, and -20, which document this inspection effort, indicate that the recertification program was satisfactorily completed.

The TRT determined that extensive engineering evalutions and tests had been conducted by TDI, the Owners' Group, and TUEC. The RIV inspectors reviewed and witnessed the satisfactory testing of the Unit 1 EDG. Therefore, if any damage did occur, it had been corrected before preoperational testing.

5. Conclusion and Staff Positions: The TRT found documented evidence supporting the alleger's concerns about damage to the EDGs in the four NCRs listed above and concluded that this allegation had potential safety significance and generic implications. However, since appropriate corrective action was taken and documented, the TRT concludes that this damage no longer exists. In addition, any damage affecting the reliability and operation of the EDGs that was not documented would have been discovered and corrected during a comprehensive recertification program undertaken by TUEC. The TRT review of TUEC and NRC Region IV documents indicates satisfactory completion of the above retests. The TRT review also confirms that the EDGs will perform in accordance with design.

   Accordingly, the TRT finds that this allegation no longer has either safety significance or generic implications.

   The TRT interviewed the alleger and provided the above findings and conclusions. The alleger indicated that his concerns were resolved.

6. Actions Required: None.

1. Allegation Category: Miscellaneous 11, Polar Crane Shimming

2. Allegation Number: AM-15, AM-16

3. Characterization: It is alleged that the shimming of the Unit 1 polar crane rail system supports was improper and that the polar crane system is improperly installed.

4. Assessment of Safety Significance: The NRC Technical Review Team (TRT) tried to contact the alleger during its inspection to learn more details about the allegation. On August 14, 1984, two TRT members visually inspected the shims from the polar crane. During the first 180° rotation of the crane, the TRT members stood on the platform above the operator's booth to view the radial restraint brackets and the seismic restraint brackets. Several brackets at different locations appeared to have gaps in excess of 1/16 inch. However, this only confirmed what had been previously observed by an NRC Region IV Resident Inspector and was documented in NRC Inspection Report (IR) No. 50-445/84-08, which required corrective action which was not yet completed.

   The TRT then moved below the operator's booth to view the polar crane rail system from another vantage point. The TRT observed the shims used for shimming 28 crane girder to girder-support brackets. During this 180° rotation, the TRT observed large gaps, particularly on the inside edge (looking from the inside of the Containment Building to the outside).

   The TRT met with the Texas Utilities Electric Company (TUEC) project civil engineer, the Brown & Root (B&R) project control manager, the B&R subcontracts supervisor, and a representative from Chicago Bridge and Iron (CB&I) to determine the gap-tolerance specification between bearing plate "A" (Dwg. 2323-S1-0515, Revision 4) and the girder to girder-support bracket. Neither Gibbs & Hill (G&H) specification SS-14 nor the Crane Manufacturers Association of America Manual (CMAA-70) addressed this issue. The meeting failed to produce a specific answer; however, copies of two letters related to the issues were provided. The first, a B&R letter (No. BRF-7404), dated November 8, 1977, contained the as-built measurements of gaps at all shim locations and a request for G&H to evaluate this information and provide direction. At 28 locations, the as-built drawings showed gaps that ranged up to 0.581 inch. In the second letter, G&H (GHF-2207, dated November 28, 1977) responded as follows:

   > Girder Seat Connections
   >
   > These seated connections will not require shimming since the area in bearing is at least the width of the bottom flange of the crane girder. The gap dimensions indicated in the Brown & Root survey exist only at the extreme edges of plate A, Section 3-3, Dwg. 2323-S1-0515, Revision 4.

   The TRT noted that the bottom flange of the girder referenced in the G&H letter (the bearing surface) is 20 inches wide.

   On August 30, 1984, an NRC inspector, accompanied by a TUEC quality control (QC) inspector, inspected the 28 crane girder to girder-support bracket shims. Nine girders, identified as A7-6 right-end (RE), A7-8 (RE), A7-12 (RE), A7-14 (RE), A7-18 left-end (LE), A7-19 (RE), A7-20 (RE), A7-24 (RE), and A7-25 (RE), had gaps in excess of 1/16-inch extending under the bottom flange. This observation invalidated the G&H assumption of 20 inches of bearing surface.

The TRT closely observed girder A7-20 (RE) as the crane wheels passed directly over the support bracket and saw no visible compression (closure) of the gap. In addition, a visual inspection of the complete rail system revealed that the rail has moved or is moving circumferentially, as indicated by the fact that some of the 1-inch-diameter stabilizing rods are bent from the force of this movement. The 3/8-inch designed gap between the ends of the rail section also varied from 0.000-inch to 0.875-inch, when measured at the inside edge of the rail. In addition, three of the rail-to-rail ground wires and two Cadwelds were broken, and at least two rail shim plates had partially worked out from under the rail.

The TRT interviewed the polar crane operator and asked if he knew of any existing problems with the crane or its operation. He replied that the crane operates satisfactorily and has experienced no apparent problems. He also stated there are no "dead spots" (i.e., no loss of electrical energy at spots) in the bus bars.

The TRT found additional shimming problems and additional types of problems, described above, that had not previously been identified. These deficiencies appear to be safety significant and generic.

5. Conclusion and Staff Positions: Based on the above inspections, the TRT concludes that this allegation is substantiated and is potentially safety significant. The problem with shimming and inspection of safety-related work was first identified in 1982. Because problems still existed in 1984, this matter appears to be generic.

   On November 8, 1984, the TRT interviewed the alleger to provide the above findings and conclusions. The alleger stated that his concerns were resolved.

6. Actions Required: TUEC shall inspect the polar crane rail girder seat connections for the presence of gaps which reduce the bearing surface to less than the width of the bottom flange.

   TUEC shall perform an analysis which will determine whether existing gaps are acceptable or if corrective actions are required. TUEC shall determine if additional rail movement is occurring and, if so, provide an evaluation of safety significance and the need for corrective action.

   TUEC shall perform a general inspection of the polar crane rail and the rail support system, correct identified deficiencies of safety significance, and provide an assessment of the adequacy of existing maintenance and/or surveillance programs.

Note: The gaps in the seismic restraints were the subject of NRC Inspection Reports 50-445/82-11, 50-446/82-10, and 50-445/84-08; violations were issued in each report. Although these matters may have been evaluated and a response made to the referenced violations, TUEC shall consider this matter as a part of the inspection of the polar crane system.

1. Allegation Category Miscellaneous 12, Welding of Lifting Lugs onto Tornado Missile Barrier Doors

2. Allegation Number: AM-17

3. Characterization: It is alleged that deficient welds on a missile barrier door were accepted.

4. Assessment of Safety Significance: The NRC Technical Review Team (TRT) tried unsuccessfully to contact the alleger during its inspection to learn more details about the allegation. The TRT identified the location of the doors from information provided by Texas Utilities Electric Company (TUEC) quality assurance personnel who were present at the time of the allegation, and from a review of the Atomic Safety and Licensing Board (ASLB) hearing record on intimidation and harassment. The door referred to in this allegation is the tornado missile barrier located at ground level on the west side of the Unit 1 diesel generator room. The alleged deficient welds are 18 double-grooved welds that attach the lifting lugs to the three missile barriers. These welds were terminated by wrapping the weld around the end of the lug, a practice questioned by the alleger.

   The TRT learned that the alleger mistakenly believed that neither the welds which were made nor the wraparounds terminating the welds were allowed by the weld symbol specified on the drawing. The alleger thought that a lifting lug which was welded to the flat side of the missle barrier steel door (using a double-grove T-weld joint) should have been indicated with a weld symbol showing a double groove on each side of the lifting lug and with a weld symbol at the end of the lug showing a fillet weld where the runoff occurred. The Brown & Root (B&R) inspectors who actually performed the inspections did not interpret it as the alleger did and accepted the 18 disputed welds. The TRT reviewed the B&R inspection reports, which indicated that welds were performed in accordance with Welding Procedure Specification (WPS) 10046, Rev. 9, and with American Welding Society (AWS) code requirements.

   The TRT also reviewed the construction traveler and the inspection reports for the shop fabrication and field-fitting of the missile barriers and found that, although rework occurred, no rework was done on the lifting lug welds.

   The TRT reviewed the inspection procedure, weld procedure, and inspection reports referred to in the traveler for applicability and compliance with AWS Code D1-1, Sections 4.6.1 and 4.6.2, to determine if the code allowed wrapping the weld around the end of the lug. These sections of the code state:

   > 4.6.1 Groove welds shall be terminated at the ends of a joint in a manner that will ensure sound welds. Whenever possible, this shall be done by the use of extension bars or run-off plates.

4.6.2 In building construction, extension bars or run-off plates need not be removed unless required by the Engineer.

The TRT inspected the four tornado missile barriers (east and west of the diesel generator room, Units 1 and 2). except where the missile barriers had been removed in Unit 1. In Unit 2 (east side), one segment of one missile barrier was in place for trial fitting; two segments were being fabricated on location. The TRT visually inspected the lug welds on the tornado missile barrier segments and determined that the welds showed very good workmanship and that wraparound on the ends of the lugs was both minimal and acceptable.

The TRT found that welding symbols had been correctly interpreted and that all of the welding described above, which included wrap around to terminate the welds, had been correctly done.

5. <u>Conclusion and Staff Positions</u>. The TRT found no errors in design or interpretation of weld symbols or any poor workmanship on welds of the lifting lugs on the tornado missile barriers. In addition, the TRT determined that both the welding and the inspection of doors was done in accordance with specified procedures. The lugs function only for lifting the massive missile barrier doors and would have little or nothing to do with protecting safety-related equipment from missiles. Accordingly, the TRT concludes that this allegation has neither safety significance nor generic implications.

   The TRT will provide the above findings and conclusions to the alleger by letter.

6. <u>Actions Required</u>: None.

1. Allegation Category: Miscellaneous 13, Welding of Pipe Supports in Safeguards Tunnel

2. Allegation Number: AM-18

3. Characterization: It is alleged that the tube steel used to fabricate supports by welding it to baseplates in the Unit 1 safeguards "796 yard tunnel" was cut at the wrong angle, resulting in too large a gap between the tube and baseplate.

4. Assessment of Safety Significance: The NRC Technical Review Team (TRT) visually observed the entire "796 yard tunnel" to determine how many supports were used, and found there were several hundred tube-steel-to-baseplate weldments/installations. A member of the TRT also attempted to contact the alleger to obtain additional information to determine the approximate location, size, and configuration of the subject tube-steel-to-baseplate weldments, because without this information indiscriminate destructive testing of the installed supports would be necessary in order to identify the location of the alleged gap. However, the alleger refused to communicate with the TRT.

   The TRT visually observed the "796 yard tunnel" to identify any condition that would show improper installation or welding on the 796-foot, 6-inch elevation, and the three, short 800-foot-elevation tunnels. Subsequently, the TRT randomly selected typical pipe supports and five hanger inspection reports (DD-1-16-025-Y33R, DD-1-16-024-Y33R, SI-1-031-041-532K, SW-1-17-716-Y33K, and AF-1-002-033-Y33K) and reviewed them to determine if the documentation of the inspections required during installation was correct. The inspection reports indicated that the supports were correctly installed in accordance with Brown & Root procedures CP-CPM-6.9E, "Pipe Fabrication and Installation," Revision 7; CP-CPM-6.9F, "Fabrication and Installation of Component Supports," Revision 0; and CP-CPM-7.1G, "Piping Supports," Revision 0.

   The TRT requested that a Brown & Root inspector take copies of the inspection reports for the five pipe hangers to the field and repeat those steps in the inspection which could be repeated; however, no root gaps could be inspected because all welding had been completed. Inspection included dimensional checks, measurement of welds, checks for proper anchoring, and visual inspection of welds. The TRT found no major discrepancies between the inspection reports and the field conditions. The recheck of the five inspection packages also showed that the as-installed and as-inspected conditions agreed, which indicated that both workmanship and inspection on the five hangers were adequate. Most of the rework required by the nonconformance reports (NCRs) found in the inspection packages consisted of filling in undersized fillet welds, although some rework was initiated by design change authorization (DCA).

   The TRT learned that from July to September 1984, the NRC Region IV (RIV) inspectors reviewed a portion of the Unit 1 auxiliary feedwater system while performing inspection 50-445/84-26. The RIV inspectors paid specific attention to two water lines which were connected to the condensate water storage tank (CP-AFATCS-01) and were located in the safeguards tunnel.

They also inspected the pipe support in this area to the "as-built" vendor-certified drawing (VCD), and included critical dimensions of support members, weld size and type, support location, clearances, baseplates, workmanship, anchor bolt type and placement in their inspection. The RIV inspectors also reviewed the document package for the support.

The RIV Inspectors inspected the following 10-inch supply line (AF-1-01-152-3) pipe supports/restraints identified on BRHL-AF-1-YD-002, and found no deficiencies or deviations.

AF-1-001-020-Y33R	Wall-mounted, single snubber
AF-1-001-021-Y33K	Floor-mounted, double snubber
AF-1-001-025-Y33R	Wall-mounted, rigid strut
AF-1-001-028-Y43K	Wall-mounted, double snubber
AF-1-001-030-Y33R	Ceiling-mounted, double strut
AF-1-001-036-Y33R	Wall-mounted, single strut

The RIV inspectors also examined the 8-inch return line (AF-1-035-152-3) to the condensate water storage tank and the following pipe supports/restraints, which are identified on BRHL-AF-1-YD-001, and found no deficiencies or deviations.

AF-1-035-001-Y33R	Wall-or ceiling-mounted seismic pipe restraint
AF-1-035-003-Y33R	Wall-or ceiling-mounted seismic pipe support
AF-1-035-032-Y33R	Wall-mounted seismic sway strut
AF-1-035-034-Y33R	Wall-mounted seismic sway strut
AF-1-035-035-Y33K	Wall-mounted double seismic snubber
AF-1-035-037-Y33R	Wall-mounted seismic sway strut

The TRT determined that the RIV inspection identified no deficiencies such as those described by the alleger.

In assessing this allegation, the TRT attempted to inspect the alleged deficient weld root gaps. The primary responsibility for a broader and more in-depth inspection of hangers belonged to other TRT Groups and, as described above, to the Region IV report.

The TRT and a Brown & Root inspector went to the Unit 2 Safeguards Building tunnel and observed work in progress. Most of the piping had been installed, and approximately 12 pipe fitters were installing permanent hangers and snubbers on the tube steel. Fitups before welding appeared tight (less than 1/16-inch), straight, and uniform. The TRT observed no deficient fitups on the diagonal runs of the tube steel as described by the alleger.

These findings are based on the technical assessment performed by the Miscellaneous Group; some of these findings appear to differ from those made by the TRT's QA/QC Group, which evaluated the same components from a different technical perspective.

5. <u>Conclusion and Staff Positions</u>: The TRT was unable to identify any improper fitups or gaps related to installed tube to baseplate weldments. The workmanship concerning the subject supports appeared to be good. Accordingly, this allegation has neither safety significance nor generic implications.

The TRT was unable to provide the above findings and conclusions to the alleger because the alleger could not be located. Several attempts were made by letter and telephone to locate this alleger; however these attempts were unsuccessful.

6. <u>Actions Required</u>: None.

1. Allegation Category: Miscellaneous No. 14, Posting of NRC Form-3

2. Allegation Number: AM-19

3. Characterization: It is alleged that posting requirements for NRC Form-3 were not met during 1977-1982.

4. Assessment of Safety Significance: The NRC Technical Review Team (TRT) reviewed the allegation, which was made in NRC Office of Investigation (OI) document A4-83-005 dated May 20, 1984, and found no need to contact the alleger to further clarify the allegation.

   The TRT reviewed the deposition given by Robert R. Taylor dated July 17, 1984, in which this NRC inspector stated that Texas Utilities Electric Company (TUEC) had posted a memorandum or letter in 1978 (about 6 months before Taylor became the resident inspector at the plant). This letter invited any site employee to contact the NRC if they had concerns about the quality of the construction of the plant. The NRC Region IV telephone number was the point of contact. This letter was the forerunner of the NRC Form-3 which became a posting requirement in October 1982. Mr. Taylor could not recall if the Form-3 was posted in October 1982, but had prepared a report in January 1983, which documented the posting.

   The Atomic Safety and Licensing Board (ASLB) deposition of C. Tedder, H. Hollis, C. Baker, and M. Hall, dated July 18, 1984, stated that NRC Form-3 had been posted from October 1982 until the present. The bulletin boards were periodically checked to assure proper posting.

   The TRT reviewed 10 CFR Part 50 dating back to 1976, and learned that the 10 CFR Part 50.7 requirement for posting the NRC Form-3 around sites under construction was not effective until October 12, 1982. The TRT observed the locations of the 12 Comanche Peak Steam Electric Station (CPSES) project bulletin boards currently in use, as well as 5 additional bulletin boards in various work spaces. The TRT determined that Form-3 was properly posted on all bulletin boards.

   The TRT interviewed the TUEC Radiation Protection Engineer (RPE), who currently is responsible for maintaining the CPSES Unit 1 bulletin boards, and the TUEC administrative and control supervisor, who is now responsible for maintaining the 12 CPSES project bulletin boards.

   The RPE stated that TUEC designates official bulletin boards in work areas and other assembly areas and reviews them periodically to ensure compliance with posting requirements. The TUEC administrative and control supervisor also stated that the CPSES project bulletin boards are reviewed periodically to ensure compliance with posting requirements and that locations for the bulletin boards may change as construction progresses. Management has not formally assigned responsibility in writing for establishing bulletin board locations or for maintenance and periodic review; the responsibility is informally assumed.

   The TRT also telephoned the Texas Utilities Service, Inc. (TUSI) personnel manager who was responsible for maintenance of the bulletin boards during the September 1982 to October 1983 period. The TUSI personnel manager stated that approximately five bulletin boards were in place and that an additional six bulletin boards were installed during his period of responsibility. This responsibility, however, was not a formally assigned job function.

5. Conclusion and Staff Positions: Based on a review of the NRC and TUEC depositions, interviews with the RPE, the TUEC administrative and control supervisor and the TUSI personnel manager, and inspection of bulletin boards currently in place, the TRT concludes that letters were posted prior to October 1982, and that the NRC Form-3 was posted in a sufficient number of places to meet the intent of the applicable regulations after the posting requirements became effective on October 12, 1982. Formal or written assignment of responsibility for NRC Form-3 posting could strengthen TUEC's program if a policy of assigning responsibility were established. Since there was no requirement to post NRC Form-3 between 1977 and October, 1982, and the form was posted for the balance of 1982 until the present, this allegation is not substantiated.

   .he TRT will provide the above findings and conclusions to the alleger by letter.

6. Actions Required: TUEC shall formally establish in writing the assignment of responsibility for posting and maintaining NRC Form-3 in prominent locations.

1. Allegation Category: Miscellaneous 15, Drug Abuse

2. Allegation Number: AM-21

3. Characterization: In a letter dated March 7, 1984, it is alleged that there was widespread drug use and abuse at the Comanche Peak Steam Electric Station (CPSES) and that management did not give proper attention to the alleged problem.

4. Assessment of Safety Significance: The NRC Technical Review Team (TRT) reviewed background material and an NRC Region IV report of inspection pursuant to temporary instruction (TI 2596/1) that documented discussions held with representatives of Texas Utilities Electric Company (TUEC) in early 1983. This report described TUEC's company policy on use or possession of drugs and alcohol, employee assistance programs, background checks/psychological tests, supervisory and employee awareness of drug/alcohol problems, and a drug/alcohol abuse detection program.

   During its review, the TRT learned that a drug abuse prevention program had been in effect at CPSES since 1974, when work on the project began, and that it included TUEC policy statements which emphasized that any employee possessing drugs or alcohol on company property was subject to immediate discharge. As a part of the review, the TRT interviewed managers, staff, and technicians affiliated with TUEC and Brown & Root (B&R). The staff also interviewed medical laboratory personnel and law enforcement officials in the CPSES area. The topics discussed in these interviews included the following:

   - The methodology used in conducting drug investigations.
   - The techniques used during pre-employment background investigations.
   - The program for supervisors to ensure that they recognized employees with potential problems.
   - The employee assistance program for permanent TUEC employees.

   The TRT determined through interviews and a review of personnel department practices and records that B&R corporate policy required prospective employees to take a physical examination to satisfy insurance or workmen's compensation requirements; however, the urinalysis given in this physical examination did not include an analysis for drugs.

   The TRT also learned that as a precondition of employment, both at TUEC and at Burns International (the CPSES physical security contractor), any employee requiring unescorted access to secured areas had to provide a urine specimen which was analyzed for a wide range of drugs. Prospective TUEC and security contractor employees were also subjected to an in-depth background investigation. TUEC's physical security plan commits them to mandatory screening and investigation of potential employees when the reactor becomes operational; however, TUEC elected to put these policies into effect prior to reactor operation.

The TRT reviewed procedures and examined materials which revealed that, in anticipation of operation of CPSES, TUEC had initiated an indoctrination program for supervisors to aid them in recognizing unusual behavior caused by alcohol or drug abuse.

Since B&R is the constructor and will not be involved with the operation of CPSES, there was no requirement in the TUEC Safeguards Security Plan concerning B&R personnel. Therefore, the B&R employees who were using drugs would not have been detected using this screening process. However, other screening measures, including the periodic use of dogs trained to detect drugs, were used to compensate for this lack of screening.

The TRT interviewed the county sheriff and found that TUEC notified law enforcement authorities about their investigation regarding drug involvement by B&R employees and kept them advised of their findings.

The measures described above were directed by TUEC and B&R management, but despite these measures an incident occurred. The TRT learned that in early June 1984, TUEC security investigators from the corporate office in Dallas followed up on alleged onsite drug abuse by B&R employees, which reportedly occurred in the Unit 2 construction area and involved personnel from several trades and fields. TUEC began the investigation by interviewing the alleger and, using a networking approach, conducted a series of interviews with 56 workers at CPSES. Following these interviews, TUEC security requested that 39 B&R employees take polygraph tests to support statements they had made when interviewed. This information was referred to the B&R personnel office and, as a result, 33 of the 39 employees who had been implicated by the interviews terminated employment.

The TRT also interviewed the TUEC site QA manager and his special staff assistant concerning the use of drugs by quality assurance/quality control (QA/QC) personnel and its impact on safety-related work activities. They stated that in early June 1984, a TUEC investigator advised them that eight B&R QA/QC employees had been identified as being involved with drugs. They also stated that either three or four of the eight had left the project prior to the investigation, and the remainder terminated employment when the results of the investigation were referred to the B&R personnel office.

Because inspections of safety-related work made by 8 of the 39 B&R employees involved with drugs may have been inadequate, a nonconformance report (M84-01840, dated June 15, 1984) was issued. This NCR addressed items/components in every system in Units 1 and 2, as those employees identified with drug involvement had worked in all areas of Unit 1 and 2. However, B&R interoffice memorandum (IOM), dated July 18, 1984, and TUEC IOM (TUQ-2289), dated August 14, 1984, provided justification to TUEC management to exclude reinspection of the work of three B&R inspectors, either because an authorized nuclear inspector (ANI) had independently inspected this work and found no problems or because the work was not safety related. Therefore, the work of only five of the B&R inspectors involved in drugs was reinspected.

The TRT found that in response to the NCR, TUEC QA personnel developed a reinspection program to assess the adequacy of those inspections which might have been inadequate. This program involved determining the total number of inspections for each of the inspectors and selecting a statistical sampling plan from MIL-STD-105D, "Sampling Procedures and Tables for Inspection by Attributes." The sampling plan taken from MIL-STD-105D provided for a General Inspection, Level II, single, normal inspection with an acceptable quality level (AQL) of 4.0 considered to be adequate.

The TRT reviewed the results of TUEC's reinspection program and found that the reinspected items/components were randomly selected and that a valid sample was reinspected by B&R inspectors. On August 31, 1984, the TUEC QA staff engineer stated that no significant deficiencies had been identified during their reinspection effort; however, eight minor deficiencies were referred to engineering in TUEC IOM QA-0047, dated September 21, 1984. IOM QA-0047 also included a request for an evaluation of the safety implications of these minor deficiencies had they gone undetected. (The final sign off of this NCR was not completed as of January 31, 1985, pending TUEC engineering and legal review.)

Following TUEC's reinspection program, the TRT randomly selected five percent of the TUEC sample to verify the adequacy of the reinspection. (The B&R inspectors involved in drug-related activities were identified as A, B, C, D, E, F, and G.) The TRT also included the work of the two inspectors whose work had not been reinspected by TUEC/B&R personnel. One B&R inspector's work was not included in this sample because it pertained to coatings, an area which was extensively inspected and evaluated by the TRT Coatings Group. Based on this sample, the TRT determined that the inspections were adequate. In addition, the TRT found no items/components that were deficient.

The TUEC QA manager stated that although some craft personnel had been involved in drug abuse, their work was not reinspected because they were not responsible for final acceptance of their own work, but relied on in-process inspections and a final acceptance inspection made by B&R inspectors who were not involved in the drug-related incident. In addition, an ANI inspected work done by craft personnel when ASME work was involved. The TRT evaluated this position and reasoned that the sample of all inspector's work would also include a sample of craft personnel work, and if no significant deficiencies were found, their justification would be accepted.

5. Conclusion and Staff Position: Based on the above review, the TRT concludes that TUEC had performed an investigation and identified B&R personnel implicated by their refusal to take polygraph tests and their subsequent termination of employment. Although this allegation had potential safety significance and generic implications, TUEC wrote a nonconformance report which identified all work performed by the implicated B&R inspectors and reinspected by different inspectors. The reinspection identified only minor deficiencies that have been referred to engineering for final evaluation and correction. This allegation appears to have some substance.

With respect to management, the TRT concluded that TUEC and site contractor management and supervision had implemented strong measures to prevent drug use and abuse by CPSES personnel. In fact these commitments to such a program exceed current NRC requirements and standards. Therefore, there was no evidence that management did not give proper attention to the alleged problem to prevent drug use and abuse or deal with the incident that occurred.

The TRT will provide its findings and conclusions to the appropriate group and to the alleger who was involved with this allegation.

6. Actions Required: TUEC shall provide a report of findings including the final engineering analysis of the minor deficiencies.

1. Allegation Category: Miscellaneous 16, Heating, Ventilating, and Air Conditioning System

2. Allegation Number: AM-22

3. Characterization: It is alleged that Texas Utilities Electric Company (TUEC) has not analyzed the heating, ventilating, and air conditioning system (HVAC) supports for seismic loads; that all HVAC components and supports inside the Containment Building were not properly considered in regard to their treatment as missiles; that the HVAC system is not properly supported; and, that HVAC failure during a postulated accident would allow the temperatures to rise to an unacceptable level inside the Containment Building.

4. Assessment of Safety Significance: The NRC Technical Review Team (TRT) found no need to contact the alleger to further clarify the allegation. The TRT reviewed the Comanche Peak Steam Electric Station (CPSES) Final Safety Analysis Report (FSAR) to identify the HVAC's design and quality assurance requirements. FSAR Volume IV, Section 3.2, "Classification of Structures, Components and Systems," states that part of the containment ventilation system is seismic Category I; however, FSAR Volume XIV, Section 17.0, Appendix 17A, "List of Quality Assured Items," states that the containment ventilation system (which contains eight subsystems/components) is seismic Category II and nonsafety related with the exception of the containment purge exhaust ductwork, supports, debris screen, and isolation valves, which are seismic Category I. Only the isolation valves, which are safety and code class 2, are safety related and seismic Category I.

The TRT determined that the entire containment ventilation system is nonsafety related, except for the isolation valves referenced above. None of these nonsafety-related systems is necessary for the safe shutdown of the reactor or to prevent or mitigate the consequences of accidents or malfunctions in the reactor coolant pressure boundary. All components, except the containment purge exhaust, which are inside the Containment Building are seismic Category II, and need not operate during a safe shutdown earthquake (SSE) but simply are components which are not allowed to fall and damage an essential safety-related system. Therefore, the HVAC ductwork is not required to remove heat from containment. The system that removes heat from Containment Building is the containment spray system, which does not depend on HVAC duct work or HVAC supports. The allegation that temperatures would rise to an unacceptable level because of an inoperative HVAC is incorrect.

The containment purge exhaust is classified as nonsafety related and seismic Category 1, which means it is designed to continue operating during a SSE; however, this ductwork system is not essential for the safe shutdown of the plant. The containment isolation valves close on a signal of high radiation to prevent a release to the environment, as specified by 10 CFR Part 20. This system is not used to remove heat from the Containment Building either, as previously discussed. This is a containment spray system function so that the alleged high temperatures could not be caused by inoperative containment exhaust HVAC ductwork and supports.

The TRT reviewed FSAR Volume IV Section 3.5 and determined that TUEC had considered internally generated missiles inside the Containment Building. The allegation that the HVAC was not considered with respect to missiles is incorrect.

The TRT reviewed NRC Region IV (RIV) Inspection Report (IR) 50-445/83-24; 50-446/83-15 which documented a review of the allegations characterized above. It concluded that these allegations were without merit. This inspection based these conclusions on the review of the FSAR; an NRC Construction Appraisal Team report (CAT), dated April 11, 1983; and a special NRC inspection at Corporate Consulting & Development Company, LTD (CCL) the consultant responsible for HVAC design. The inspection of the consultant's analysis of design and seismic requirements, i.e., the seismic design techniques and assumptions, was acceptable.

The TRT learned that RIV IR 50-445/84-16 documented a special inspection of the reactor Containment Building. Twenty-five duct supports segments in the Unit 1 containment air circulation and cooling system were inspected. The seismic supports were inspected to assure that installations were as designed or deviations were analyzed to assure the adequacy of the support of the HVAC systems.

As a result of an interview with the alleger, additional inspection was performed and documented on December 18, 1984. The TRT randomly selected and observed various HVAC systems; the HVAC appeared to be properly supported. The HVAC inside the Containment Building was analyzed and reported in CCL seismic analysis reports dated December 18, 1981, and July 24, 1984. The latter provided the following information: (1) the duct hangers were analyzed on a hanger-by-hanger basis, (2) the analysis was based on the latest as-built drawing, and (3) the hangers were designed and analyzed as frame structures having diagonal braces or without braces, thus relying on the bending of vertical supports to support lateral loads. The latter method may have caused the allegation that the HVAC was unsupported.

5. <u>Conclusions and Staff Positions</u>: Based on the review of design requirements in the FSAR, the review of NRC inspections, and visual inspection of the HVAC systems, the TRT concludes that the allegations are based on the erroneous assumption that HVAC is required during a design basis accident. The HVAC system has been properly designed and analyzed by an independent seismic consultant and by an analysis which included consideration of vertical and lateral supports needed to meet seismic Category 1 and 2 requirements. Internally generated missiles inside containment were also analyzed. This allegation is not substantiated; therefore, it has neither safety significance nor generic implications.

   The TRT presented the above findings and conclusions to the alleger and agreed to recheck lateral supports at the alleger's request. After so doing, the TRT concluded that lateral supports were adequately considered.

6. <u>Actions Required</u>: None.

1. Allegation Category: Miscellaneous 17, Damage to Upper Internals

2. Allegation Number: AM-24

3. Characterization: It is alleged that damage occurred to the 15-foot by 2 1/2-inch stainless steel bars (subsequently determined by the NRC staff to be thermocouple columns) located in the reactor vessel upper internal structures in the Unit 1 Reactor Building at Comanche Peak Steam Electric Station (CPSES). The alleger's concern is that approximately 1 foot from the top of the stainless steel bars, two of them were bent when they were struck by either a fork lift or a crane. The alleger contends that a rope pulled by a crane was then placed around the stainless steel bars and pulled in order to straighten them. It is further alleged that no documentation was ever completed to show that this damage occurred.

4. Assessment of Safety Significance: NRC Region IV (RIV) inspected this allegation and documented the results in Inspection Report 50-445/84-08, 50-446/84-04 (July 26, 1984). Prior to this inspection (March 1984), the RIV discussed this allegation with the alleger by telephone. There was also an NRC Office of Investigation (OI) inquiry on this matter.

   The NRC Technical Review Team (TRT) reviewed OI report QA-84-016 dated April 11, 1984, the notes from the telephone conversation with the alleger, and NRC followup on Inspection Report 40-445/84-08, 50-446/84-04. In addition, the TRT visually inspected the upper internal structures of the reactor vessel and reviewed a computerized index of CPSES documentation, i.e., nonconformance reports (NCRs) and/or procedures related to damage or problems in the upper internals or reactor vessel head areas. Two documents, Westinghouse Field Deficiency Report (FDR) TBXM-10285 and Brown & Root (B&R) NCR M-11438, indicated that on October 14, 1983, the refueling crane (a bridge crane that straddles the refueling cavity) was moved without crane interlocks or a "flagman," a condition which resulted in a bent thermocouple column. Although interlocks are normally used in such a case, at the time the alleged damage occurred no specific procedure was in effect that required use of interlocks.

   The TRT learned that when the alleged damage occurred, the upper core assembly was mounted on extension legs and was stored in its designated location in the refueling cavity. The extension legs elevated the upper internals so that the thermocouple column was in the refueling crane's normal path. Each of the four thermocouple columns (tubes) is approximately 17 feet long and provides support for the incore thermocouple tubing located between the upper core internals and the reactor vessel head; the bottom of each thermocouple column is attached to the upper core assembly. The thermocouples in these columns are chromel-alumel wires that are threaded into guide tubes which penetrate the reactor vessel head through seal assemblies and terminate at the top end of the fuel assemblies. Thermocouple readings are monitored by a computer, with backup readout provided by a precision indicator from the incore instrumentation, even if the computer is not in service. These thermocouples are not required for safety. (See Final Safety Analysis Report [FSAR], Section 7.7 "Control Systems Not Required for Safety.")

Westinghouse FDR TBXM-10285 and B&R NCR M-11438 indicated that thermocouple column R-11 (ID No. 19546, Sub. 1) on the reactor vessel internal structure was bent in an area approximately 2 feet above the support tube (Item 3 on Drawing 6116E84). The support tube had no apparent damage; however, the upper section of the thermocouple column and its respective protective sleeve were approximately 1 foot off the vertical. A review of records by the TRT indicated that, following the alleged damage, the thermocouple column had been properly aligned in a perpendicular direction. The area was also visually inspected by the TRT for cracks with a ten power magnifying glass. No cracking was observed where bending occurred. In addition, resistance readings were taken on February 6, 1984, with properly calibrated resistance instrumentation, and the results were acceptable.

The TRT review of Deficiency Report TBXM 10285 and NCR M11438 indicated (1) the recommended corrective action of using a strong back and hydraulic rams to straighten the thermocouple action was reviewed and approved, (2) criteria (visual inspection, thermocouple resistance measurements) for acceptance of the corrective action were established, reviewed and approved, and (3) required QA/QC sign offs were completed.

Between February and September 1984, Texas Utilities Electric Company (TUEC) personnel demonstrated proper operation of the manipulator crane, its interlocks, and its safety features in accordance with Procedure ICP-PT-40-03, "Manipulator Crane." Testing verified that the interlocks and safety features prevented any movement which would permit damage should procedures or personnel fail to perform as required during operation or movement of the manipulator.

Both the Westinghouse FDR and the B&R NCR indicated that TUEC identified and reported the damage to the steel bars, evaluated the potential and actual damage, and straightened the thermocouple column. Subsequent evaluation by the TRT determined that the corrective action taken was appropriate and adequate.

In addition, the TRT inspected other areas, but identified no other documentation nor any physical evidence (dents, deep scratches, misalignment, or gouges) related to stainless steel or carbon steel with diameters between 1 and 4 inches on the reactor vessel head, the upper internals, or the core barrel.

5. <u>Conclusion and Staff Positions</u>: The TRT determined that the stainless steel bars (thermocouple columns) were bent by the refueling crane and corrected by the recommended action of using a strong back and hydraulic ram(s). Further review indicated that the thermocouples are not safety related, however. The TRT concludes that TUEC's reporting and corrective actions were appropriate for this type of equipment damage, and that an appropriate level of quality was applied to the corrective actions. The corrective actions taken indicate that no additional repairs or potential deleterious equipment failure should result from the bending of the thermocouple column and subsequent straightening. Accordingly, this allegation has neither safety significance nor generic implications.

On November 1, 1984, the TRT provided the above findings and conclusions to the alleger. The alleger stated that his questions or concerns were answered and he had no further concerns.

6. Actions Required: None.

1. Allegation Category: Miscellaneous 18, Broken Internal Wires in Polar Crane Festooned Cable and Crane Movement Interference

2. Allegation Number: AM-25

3. Characterization: It is alleged that internal wires were broken in the polar crane festooned cables and that the polar crane hit unspecified hangers while operating.

4. Assessment of Safety Significance: The alleger was interviewed by the NRC Technical Review Team (TRT) on August 3, 1984, to obtain additional information regarding the allegation. On August 30, 1984, the TRT and a Texas Utilities Electric Company (TUEC) quality control (QC) inspector visually examined the festooned cables. There was no visible damage on any of the cables. In addition, a review of preoperational inspection megger test data sheets revealed that all tests were satisfactory.

   The TRT visually inspected the polar crane during three rotations to determine if there were any interferences between the crane and supports or other installed items, and noted no interferences. It is possible that the alleger was referring to the problem of the uplift lugs striking the crane girder stiffener plates, which is described in nonconformance report (NCR) M-81-00064; however, this problem was resolved in accordance with DCA 11311, Rev. 1.

   On September 12, 1984, the TRT, accompanied by a Brown & Root (B&R) QC electrical inspector and electrician, opened the two electrical junction boxes that feed the festoons on the polar crane walkway. The inspector visually inspected all of the wires in both boxes and found no broken or non-terminated wires. The TRT asked the operator of the crane about problems with the crane, specifically asking if the limit switches cut out properly. The crane operator then demonstrated the operation of the bridge crane, running it until the limit switch cut out and a signal light indicated that it cut out. Again, he stated that there were no problems with the crane.

   The TRT determined from a B&R QC inspection that no records or nonconformance reports existed which may have documented the alleged defective festooned cables because they were classified as nonsafety related. The TRT found only the records for megger testing which were previously discussed.

5. Conclusion and Staff Positions: The TRT found no damaged festooned cables. However, the polar crane uplift lugs did strike the crane girder stiffener plates and this had potential safety significance and generic implications. The TRT did find an NCR documenting the damaged plates corrective action, and the TRT verified that corrective action was taken, i.e., the crane now operates without such interference. Based on a review of applicable documentation, examination of the polar crane cables and wiring on the polar crane walkway, and interviews with the crane operator, the TRT concludes that this allegation was substantiated; however, appropriate corrective action was taken. Accordingly, this allegation has neither safety significance nor generic implications.

On November 1, 1984, the IRT provided the above findings and conclusions to the alleger. The alleger was satisfied with the findings and had no further concern regarding this matter. However, during this interview the alleger brought up a commercial concern regarding premature replacement of the cable. Although this is not related to a safety issue, the NRC's Senior Resident Inspector stated he would review the matter during a future routine inspection.

6. Actions Required: None.

1. Allegation Category: Miscellaneous 19, Chloride Contamination of Radwaste System Piping

2. Allegation Number: AM-30

3. Characterization: It is alleged that workers habitually urinated on stainless steel pipe located in the radwaste system.

4. Assessment of Safety Significance: The NRC Technical Review Team (TRT) found no need to contact the alleger to further clarify the allegation. The TRT reviewed background information pertinent to this allegation and searched the Texas Utilities Electric Company's (TUEC) quality assurance records relating to piping cleanliness. Nonconformance report (NCR) M-82-00305 described an instance where piping in the radwaste area (above the waste monitor tanks in Room 2) was "contaminated by unknown liquid substances of unspecified chemical composition." The NCR stated that hold tags were applied and lines 2 WP-X-218-151-R5 and 2 WP-X-208-151-R5, located in the Unit 1 Auxiliary Building (elevation 790 feet of Gibbs & Hill drawing 2323-A1-0507, Revision 9), were subsequently cleaned and swipe-tested according to Procedure QI-QP-11.1-65 to assure that the surfaces were free of chlorides and fluorides. TUEC verified corrective action and closed this NCR on May 27, 1982.

   The TRT discussed this NCR with TUEC quality and engineering personnel, who stated that the NCR quoted a report made by a QC inspector who witnessed a worker urinating on the piping. TUEC personnel further stated that they knew of no other similar instances; however, all safety-related stainless steel piping surfaces (outside) are routinely cleaned prior to final turnover.

   The safety significance related to chlorides, a chemical present in human urine, on stainless steel surfaces depends on the service conditions and residual stresses that may be present. If excessive stress (near the yield strength) and chloride contamination are present, the alloy may fail because of stress corrosion cracking. Since the alloy is expected to operate with a design load applied, it is necessary to ensure that chlorides are not present. In this case, the piping was cleaned to remove any chlorides that could have been deposited by urine. Thermal insulation is applied after cleaning and this protects safety-related piping (necessary for safe shutdown) from further contamination. The radwaste piping above the radwaste tanks is nonsafety related and is not needed for the safe shutdown of the plant.

   On August 29, 1984, the TRT inspected the radwaste areas (Rooms 179, 184, and 185) and found them locked and access to them controlled. Housekeeping appeared to be excellent, and the TRT detected no odors which might indicate that the area was further contaminated. The number of craft personnel who work in the Unit 1 buildings has been limited for several months, as compared to earlier periods, because this unit is virtually completed. Because limited work is in progress and personnel access controls are in place, it appears unlikely that other similar incidents occurred after cleaning.

The TRT randomly selected safety-related piping (lines RC-1-052-2501-R-1, 27.5 ID and 3/4-MS-1-194-1501-2) and reviewed the records to determine if the piping had been cleaned and swipe-tested in accordance with Gibbs & Hill Inc. Specification 2323-MS-100, Revision 8, and Brown & Root Inc. Procedures CP-QP-11.12, Revision 16 and QI-QP-11.1-65, Revision 4. Surface contamination reports J479, J492, and J497 document test results that show both the chloride and fluoride content are below the maximum specified limit of 0.0015 mg/dm. In addition, Region IV inspectors observed the external cleanliness of the reactor coolant system piping as part of their May 14 through June 20, 1984, inspection (documented in RIV Inspection Report 50-445/84-16) and identified no deviations or violations of requirements.

The TRT found no evidence to support that this incident occurred in any other area. During construction, toilet facilities are not always close to each work area; therefore, workers do sometimes urinate in unauthorized areas. However, the evidence indicates that all safety-related piping is cleaned and tested before being placed into service, eliminating potential contamination.

5. <u>Conclusion and Staff Positions</u>: The TRT found that an NCR was written on radwaste piping because a QC inspector saw a worker urinating on this piping. This allegation had potential safety significance and generic implications because the incident may have involved safety-related piping. The radwaste piping which was contaminated was subsequently cleaned. The TRT found no other instances where this happened; however, TUEC's procedures for maintaining chloride and fluoride surface contamination levels below specified limits appear to be acceptable, were followed, and will eliminate the contamination of critical safety-related piping, whether the incident was isolated or habitual. Moreover, the radwaste piping is nonsafety related. Accordingly, this allegation has neither safety significance nor generic implications.

   The TRT will provide written feedback to the alleger describing its findings and conclusions.

6. <u>Actions Required</u>: None.

1. Allegation Category: Miscellaneous 20, No Procedures or Guidance Provided for Rigging and Handling Large Components/Equipment

2. Allegation Number: AM-23(a)

3. Characterization: An NRC Region IV Resident Inspector identified a violation as a result of a discussion with a craft person who stated that he had not received instructions about how to rig and handle a large motor-operated valve.

4. Assessment of Safety Significance: The NRC Technical Review Team (TRT) found no need to contact the alleger to further clarify the allegation. The TRT reviewed NRC Inspection Report 50-445/79-27, 50-446/79-26 and its corresponding Notice of Violation (NOV). The TRT also reviewed the Texas Utilities Electric Company (TUEC) response to these documents (TXX-3080, dated December 18, 1979), which stated that the subject valve was not mishandled, nor was it damaged. The engineering organization had not, however, reviewed specific vendor rigging or handling recommendations or noted the procedures for loads exceeding 2000 pounds. An NRC followup inspection verified that Brown & Root (B&R) Procedures CP-CPM-6.3, 35-1195-CCP-24, 35-1195-ACP-3, and QI-QAP-13.1-1 were reviewed by TUEC and revised appropriately. NRC Inspection Report 50-445/80-18, 50-446/80-18 (dated September 19, 1980) documented corrective action during the followup inspection.

   The TRT interviewed TUEC's Rigging Craft Superintendent, Assistant Mechanical Superintendent, and Senior Staff Engineer. They stated that the revised procedures (specifically, CCP-2A, Revision 4, "Rigging"; CP-CPM-6.3, Revision 10, "Preparation, Approval, and Control of Operation Travelers"; and, CP-CPM-6.9, Revision 2, "General Piping Procedure") adequately controlled heavy lifts of equipment and components. Nonconformance report (NCR) M-2128 documented the problem which was identified as a violation, and the appropriate site personnel reviewed the NRC inspection report and concurred with the corrective action. In addition, the TRT independently reviewed the revised procedures for the control of heavy lifts of equipment and found the control of rigging and handling to be acceptable for loads less than or exceeding 2000 pounds.

5. Conclusion and Staff Positions: The TRT determined that Region IV (RIV) confirmed that the craftperson's stated need for better instructions was correct and confirmed followup inspection by the RIV inspector to verify that corrective action was accomplished in accordance with TUEC letter TXX-3080 (December 18, 1979). The TRT concludes that the failure to provide proper instructions for rigging and handling heavy loads is safety significant and has generic implications; however, corrective action was taken. No evidence of further inadequacies in this area was found; consequently the allegation requires no further action.

   The TRT tried to provide the above findings and conclusions to the alleger; however, the alleger's identification is unknown.

6. Actions Required: None.

Attachment 3

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SEP 18 1984

Dockets: 50-445
50-446

Texas Utilities Electric Company
Attn: M. D. Spence, President, TUGCO
Skyway Tower
400 North Olive Street
Lock Box 81
Dallas, Texas 75201

Dear Mr. Spence:

SUBJECT: COMANCHE PEAK REVIEW

On July 9, 1984, the staff began an intensive onsite effort designed to complete a portion of the reviews necessary for the staff to reach its decision regarding the licensing of Comanche Peak Unit 1. The onsite effort covered a number of areas, including allegations of improper construction practices at the facility.

The NRC assembled a Technical Review Team (TRT) responsible for evaluating most of the technical issues at Comanche Peak, including allegations. The TRT has recently identified a number of items that have potential safety implications for which we require additional information. These items are listed in the enclosure to this letter. Further background information regarding these issues will be published in a Supplement to a Safety Evaluation Report (SSER), which will document the overall TRT's assessment of the significance of the issues examined.

The items in the enclosure to this letter, which are in the general areas of electrical/instrumentation, civil/structural and test programs, cover only a portion of the TRT's effort. The TRT evaluation of items in the areas of mechanical, QA/QC, and coatings, and its consideration of the programmatic implications of these findings, are still is progress. A summary of these issues will be provided to you at a later date.

You are requested to submit additional information to the NRC, in writing, including a program and schedule for completing a detailed and thorough assessment of the issues identified. This program plan and its implementation will be evaluated by the staff before NRC considers the issuance of an operating license for Comanche Peak, Unit 1. The program plan should address the root cause of each problem identified and its generic implications on safety-related systems, programs, or areas. The collective significance of these deficiencies should also be addressed. Your program plan should also include the proposed TUGCO action to assure that such problems will be precluded from occurring in the future.

This request is submitted to you in keeping with the NRC practice of promptly notifying applicants of outstanding information/evaluation needs that could potentially affect the safe operation of their plant. Further requests for additional information of this nature will be made, if necessary, as the activities of the TRT progress.

Sincerely,

Darrell G. Eisenhut, Director
Division of Licensing, NRR

Enclosure:
As stated

cc w/enclosure
See next page

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ENCLOSURE 1

REQUEST FOR ADDITIONAL INFORMATION

I. Electrical/Instrumentation Area

a. Electrical Cable Terminations

The Technical Review Team (TRT) inspected random samples of safety-related terminations, butt splices inside panels, and vendor-installed terminal lugs in General Electric (GE) motor control centers, and reviewed documentation relative to the installations.

1. The TRT found a lack of awareness on the part of quality control (QC) electrical inspectors to document in the inspection reports when the installation of the "nuclear heat-shrinkable cable insulation sleeves" was required to be witnessed.

   Accordingly, TUEC shall clarify procedural requirements and provide additional inspector training with respect to the areas in which nuclear heat-shrinkable sleeves are required on splices and assure that such sleeves are installed where required.

2. The TRT found inspection reports that did not indicate that the required witnessing of splice installation was done. Examples are as follows:

IR ET-1-0005393	IR ET-1-0005396
IR ET-1-0005394	IR ET-1-0006776
IR ET-1-0005395	IR ET-1-0014790

   Accordingly, TUEC will assure that all QC inspections requiring witnessing for butt splices have been performed and properly documented; and verify that all butt splices are properly identified on the appropriate drawings and are physically identified within the appropriate panels.

3. The TRT found a lack of splice qualification requirements and provisions in the installation procedures to verify the operability of those circuits for which splices were being used.

   Accordingly, TUEC shall develop adequate installation/inspection procedures to assure that the wiring splicing materials are qualified for the appropriate service conditions, and that splices are not located adjacent to each other.

4. Selected cable terminations were found that did not agree with their locations on drawings. Examples are as follows:

Panel CP1-ECPRCB-14, Cable E0139880
Panel CP1-ECPRTC-16, Cable E0110040
Panel CP1-ECPRTC-16, Cable E0118262
Panel CP1-ECPRTC-27, Cable EG104796
Panel CPX-ECPRCV-01, Cable EG021856
Panel CP1-ECPRCB-02, Cable NK139853 (nonsafety)

Accordingly, TUEC shall reinspect all safety-related and associated terminations in the control room panels and in the termination cabinets in the cable spreading room to verify that their locations are accurately depicted on drawings. Should the results of this reinspection reveal an unacceptable level of nonconformance to drawings, the scope of this reinspection effort shall be expanded to include all safety-related and associated terminations at CPSES.

5. The TRT found cases where nonconformance reports (NCRs) concerning vendor-installed terminal lugs in GE motor control centers had been improperly closed. Examples are NCR Nos. E-84-01066 through NCR E-84-01076, inclusive.

   Accordingly, TUEC shall reevaluate and redisposition all NCRs related to vendor-installed terminal lugs in GE motor control centers.

b. <u>Electrical Equipment Separation</u>

The TRT reviewed the separation criteria between separate cables, trays and conduits in the main control room and cable spreading room in Unit 1, and the compatibility of the electrical erection specifications with regulatory requirements. The TRT reviewed documentation and inspected random samples of separation between safety-related cables, trays and conduits and between them and nonsafety-related cables, trays and conduits.

1. In numerous cases, safety-related cables within flexible conduits inside main control room panels did not meet minimum separation requirements. Examples are as follows:

   Panel CP1-EC-PRCB-02
   Panel CP1-EC-PRCB-07
   Panel CP1-EC-PRCP-06
   Panel CP1-EC-PRCB-08
   Panel CP1-EC-PRCB-09

   Accordingly, TUEC shall reinspect all panels at CPSES, in addition to those in the main control room for Unit 1, that contain redundant safety-related cables within conduits, or safety and non-safety related cables within conduits, and either correct each violation of the separation criteria, or demonstrate by analysis the acceptability of the conduit as a barrier for each case where the minimum separation is not met.

2. In several cases, separate safety and nonsafety-related cables and safety and nonsafety-related cables within flexible conduits inside main control room panels did not meet minimum separation requirements (Table 1 identifies examples of these cases). No evidence was found that justified the lack of separation.

   Accordingly, TUEC shall reinspect all panels at CPSES, in addition to those in the main control room of Unit 1, and either correct each violation of the separation criteria concerning separate cables and cables within flexible conduits, or demonstrate by analysis the adequacy of the flexible conduit as a barrier.

3. The TRT found that the existing TUEC analysis substantiating the adequacy of the criteria for separation between conduits and cable trays had not been reviewed by the NRC staff.

   Accordingly, TUEC shall submit the analysis that substantiates the acceptability of the criteria stated in the electrical erection specifications governing the separation between independent conduits and cable trays.

4. The TRT found two minor violations of the separation criteria inside panels CP1-EC-PRCB-09 and CP1-EC-PRCB-03 concerning a barrier that had been removed and redundant field wiring not meeting minimum separation. The devices involved with the barrier were FI-2456A, PI-2453A, PI-2475A, and IT2450, associated with Train A; and FI-2457A, PI-2454A, PI-2476A, and IT-2451, associated with Train B. The field wiring was associated with devices HS-5423 of Train B and HS-5574, nonsafety-related.

   Accordingly, TUEC shall correct two minor violations of the separation criteria inside panels CP1-EC-PRCB-09 and CP1-EC-PRCP-03 concerning a barrier that had been removed and redundant field wiring not meeting minimum separation.

Table 1

Examples of Cases of Safety or Nonsafety-Related Cables
In Contact With Other Safety-Related Cables Within Conduits in Control Room
Panels

1. Control Panel CP1-EC-PRCB-02 - Containment Spray System

Cable No.	Train	Related Instrument
EG139373	B (green)	Undetermined
E0139010	A (orange)	Undetermined

2. Control Panel CP1-EC-PRCB-07 - Reactor Control System

Cable No.	Train	Related Instrument
EG139383	B (green)	Reactor manual trip switch
E0139311	A (orange)	Undetermined

3. Control Panel CP1-EC-PRCP-06 - Chemical & Volume Control System

Cable No.	Train	Related Instrument
EG139335	B (green)	LCV-112C
E0139301	A (orange)	Undetermined

4. Control Panel CP1-EC-PRCB-09 - Auxiliary Feedwater Control System

Cable No.	Train	Related Instrument
E0139753	A (orange)	FK-2453A
E0139754	A (orange)	FK-2453B
E0139756	B (green)	FK-2454A
EG139288	B (green)	FK-2454B

c. Electrical Conduit Supports

The TRT examined the nonsafety-related conduit support installation in selected seismic Category I areas of the plant. The support installation for non-safety related conduits less than or equal to 2 inches was inconsistent with seismic requirements and no evidence could be found that substantiated the adequacy of the installation for nonsafety-related conduit of any size. According to Regulatory Guide 1.29 and FSAR Section 3.7B.2.8, the seismic Category II and nonseismic items should be designed in such a way that their failure would not adversely affect the function of safety-related components or cause injury to plant personnel.

Accordingly, TUEC shall propose a program that assures the adequacy of the seismic support system installation for nonsafety-related conduit in all seismic Category I areas of the plant as follows:

1. Provide the results of seismic analysis which demonstrate that all nonsafety-related conduits and their support systems, satisfy the provisions of Regulatory Guide 1.29 and FSAR Section 3.7B.2.8.

2. Verify that nonsafety-related conduits less than or equal to 2 inches in diameter, not installed in accordance with the requirements of Regulatory Guide 1.29, satisfy applicable design requirements.

d. Electrical QC Inspector Training/Qualifications

The TRT examined electrical QC inspector training and certification files, and requirements for personnel testing, on-the-job training, and recertification. The TRT also interviewed selected electrical QA/QC personnel.

1. The TRT found a lack of supportive documentation regarding personnel qualifications in the training and certification files, as required by procedures and regulatory requirements. Also, the TRT found a lack of documentation for assuring that the requirements for electrical QC inspector recertification were being met. Specific examples are:

   - One case of no documentation of a high school diploma or General Equivalency Diploma.

- One case of no documentation to waive the remaining 2 months of the required 1 year experience.

- One case where a QC technician had not passed the required color vision examination administered by a professional eye specialist. A makeup test using colored pencils was administered by a QC supervisor, was passed, and then a waiver was given.

- Two cases where the experience requirements to become a Level 1 technician were only marginally met.

- One case of no documentation in the training and certification files substantiating that the person met the experience requirements.

Accordingly, TUEC shall review all the electrical QC inspector training, qualification, certification and recertification files against the project requirements and provide the information in such a form that each requirement is clearly shown to have been met by each inspector. If an inspector is found to not meet the training, qualification, certification, or recertification requirements, TUEC shall then review the records to determine the adequacy of inspections made by the unqualified individuals and provide a statement on the impact of the deficiencies noted on the safety of the project.

2. The TRT found a lack of guidelines and procedural requirements for the testing and certifying of electrical QC inspectors. Specifically, it was found that:

   - No time limit or additional training requirements existed between a failed test and retest.

   - No controls existed to assure that the same test would not be given if an individual previously failed that test.

   - No consistency existed in test scoring.

   - No guidelines or procedures were available to control the disqualification of questions from the test.

   - No program was available for establishing new tests (except when procedures changed). The same tests had been utilized for the last 2 years.

   Accordingly, TUEC shall develop a testing program for electrical QC inspectors which provides adequate administrative guidelines, procedural requirements and test flexibility to assure that suitable proficiency is achieved and maintained.

The deficiencies identified with the electrical QC inspections have generic implications to other construction disciplines. The implications of these findings will be further assessed as part of the overall programmatic review of QC inspector training and qualification and the results of this review will be reported under the QA/QC category on "Training and Qualification."

II. Civil/Structural Area

a. Unable to Justify Reinforcing Steel Omitted in the Reactor Cavity

The TRT investigated a documented occurrence in which reinforcing steel was omitted from a Unit 1 reactor cavity concrete placement between the 812-foot and 819-foot ½-inch elevations. This reinforcement was installed and inspected according to drawing 2323-S1-0572, Revision 2. However, after the concrete was placed, Revision 3 to the drawing was issued showing a substantial increase in reinforcing steel over that which was installed. Gibbs & Hill Engineering was informed of the omission by Brown & Root Nonconformance Report CP-77-6. Gibbs & Hill Engineering replied that the omission in no way impaired the structural integrity of the structure. Nevertheless, the additional reinforcing steel was added as a precaution against cracking which might occur in the vicinity of the neutron detector slots should a loss of coolant accident (LOCA) occur. A portion of the omitted reinforcing steel was also placed in the next concrete lift above the 819-foot ½-inch level. This was done to partially compensate for the reinforcing steel omitted in the previous concrete lift and to minimize the overall area potentially subject to cracking.

The TRT requested documentation indicating that an analysis was performed supporting the Gibbs & Hill conclusion. The TRT was subsequently informed that an analysis had not been performed. Therefore, the TRT cannot determine the safety significance of this issue until an analysis is performed verifying the adequacy of the reinforcing steel as installed.

Accordingly, TUEC shall provide an analysis of the as-built condition of the Unit 1 reactor cavity that verifies the adequacy of the reinforcing steel between the 812-foot and 819-foot ½-inch elevations. The analysis shall consider all required load combinations.

b. Falsification of Concrete Compression Strength Test Results

The TRT investigated allegations that concrete strength tests were falsified. The TRT reviewed an NRC Region IV investigation (IE Report No. 50-445/79-09; 50-446/79-09) of this matter that included interviews with fifteen individuals. Of these, only the alleger and one other individual stated they thought that falsification occurred, but they did not know when or by whom. The TRT also reviewed slump and air entrainment test results of concrete placed during the period the alleger was employed (January 1976 to February 1977) and did not find any apparent variation in the uniformity of the parameters for concrete placed during this period. Although the uniformity of the concrete placed appears to minimize the likelihood that low concrete strengths were obtained, other allegations were raised concerning the falsification of records associated with slump and air content tests. The Region IV staff addressed these allegations by assuming that concrete strength test results were adequate. Furthermore, a number of other allegations dealing with concrete placement problems (such as deficient aggregate grading and concrete in the mixer too long) were also resolved by assuming that concrete strength test results were adequate. The TRT agrees with Region IV that, while the preponderance of evidence suggests that falsification of results did not take place, the matter cannot be resolved completely on the basis of concrete strength test results, especially if there is any doubt about whether they may have been falsified. Due to the importance of the concrete strength test results, the TRT believes that additional action by TUEC is necessary to provide confirmatory evidence that the reported concrete strength test results are indeed representative of the strength of the concrete installed in the Category I concrete structures.

Accordingly, TUEC shall determine areas where safety-related concrete was placed between January 1976 and February 1977, and provide a program to assure acceptable concrete strength. The program shall include tests such as the use of random Schmidt hammer tests on the concrete in areas where safety is critical. The program shall include a comparison of the results with the results of tests performed on concrete of the same design strength in areas where the strength of the concrete is not questioned, to determine if any significant variance in strength occurs. TUEC shall submit the program for performing these tests to the NRC for review and approval prior to performing the tests.

c. Maintenance of Air Gap Between Concrete Structures

The TRT investigated the requirements to maintain an air gap between concrete structures. Based on the review of available inspection reports and related documents, on field observations, and on discussions with TUEC engineers, the TRT cannot determine whether an adequate air gap has been provided between concrete structures. Field investigations by B&R QC inspectors indicated unsatisfactory conditions due to the presence of debris in the air gap, such as wood wedges, rocks, clumps of concrete and rotofoam. The disposition of the NCR relating to this matter states that the "field investigation reveals that most of the material has been removed." However, the TRT cannot determine from this report (NCR C-83-01067) the extent and location of the debris remaining between the structures.

Based on discussions with TUEC engineers, it is the TRT's understanding that field investigations were made but that no permanent records were maintained. In addition, it is not apparent that the permanent installation of elastic joint filler material ("rotofoam") between the Safeguards Building and the Reactor Building, and below grade for the other concrete structures, is consistent with the seismic analysis assumptions and dynamic models used to analyze the buildings, as these analyses are delineated in the Final Safety Analysis Report (FSAR). The TRT, therefore, concludes that TUEC has not adequately demonstrated compliance with FSAR Sections 3.4.1.1.1, 3.8.4.5.1, and 3.7.B.2.8, which require separation of Seismic Category I buildings to prevent seismic interaction during an earthquake.

Accordingly, TUEC shall:

1. Perform an inspection of the as-built condition to confirm that adequate separation for all seismic category I structures has been provided.

2. Provide the results of analyses which demonstrate that the presence of rotofoam and other debris between all concrete structures (as determined by inspections of the as-built conditions) does not result in any significant increase in seismic response or alter the dynamic response characteristics of the Category I structures, components and piping when compared with the results of the original analyses.

d. Seismic Design of Control Room Ceiling Elements

The TRT investigated the seismic design of the ceiling elements installed in the control room. The following matrix designates those ceiling elements present in the control room and their seismic category designation:

1. Heating, Ventilating and Air Conditioning	- Seismic Category I
2. Safety-Related Conduits	- Seismic Category I
3. Nonsafety-Related Conduits	- Seismic Category II
4. Lighting Fixtures	- Seismic Category II
5. Sloping Suspended Drywall Ceiling	- Non-Seismic
6. Acoustical Suspended Ceiling	- Non-Seismic
7. Lowered Suspended Ceiling	- Non-Seismic

According to Regulatory Guide 1.29 and FSAR Section 3.7B.2.8, the seismic Category II and nonseismic items should be designed in such a way that their failure would not adversely affect the functions of safety-related components or cause injury to operators.

For the nonseismic items (other than the sloping suspended drywall ceiling), and for nonsafety-related conduits whose diameter is 2 inches or less, the TRT could find no evidence that the possible effects of a failure of these items had been considered. In addition, the TRT determined that calculations for seismic Category II components (e.g., lighting fixtures) and the calculations for the sloping suspended drywall ceiling did not adequately reflect the rotational interaction with the nonseismic items, nor were the fundamental frequencies of the supported masses determined to assess the influence of the seismic response spectrum at the control room ceiling elevation would have on the seismic response of the ceiling elements.

Accordingly, TUEC shall provide:

1. The results of seismic analysis which demonstrate that the nonseismic items in the control room (other than the sloping suspended drywall ceiling) satisfy the provisions of Regulatory Guide 1.29 and FSAR Section 3.7B.2.8.

2. An evaluation of seismic design adequacy of support systems for the lighting fixtures (seismic Category II) and the suspended drywall ceiling (nonseismic item with modification) which accounts for pertinent floor response characteristics of the systems.

3. Verification that those items in the control room ceiling not installed in accordance with the requirements of Regulatory Guide 1.29 satisfy applicable design requirements.

4. The results of an analysis that justify the adequacy of the nonsafety-related conduit support system in the control room for conduit whose diameter is 2 inches or less.

5. The results of an analysis which demonstrate that the foregoing problems are not applicable to other Category II and nonseismic structures, systems and components elsewhere in the plant.

e. Unauthorized Cutting of Rebar in the Fuel Handling Building

The TRT investigated an alleged instance of unauthorized cutting of rebar associated with the installation of the trolley process aisle rails in the Fuel Handling Building. The claim is that during installation of 22 metal plates in January 1983, a core drill was used to drill about 10 holes approximately 9 inches deep. The TRT reviewed the reinforcement drawings for the Fuel Handling Building and determined that there were three layers of reinforcing steel in the top reinforcement layer of the slab. This reinforcement layer consisted of a No. 18 bar running in the east-west direction in the first and third layers, and a No. 11 bar running in the north-south direction on the second layer. The review also revealed that the layout of the reinforcement and the trolley rails was such that the east-west reinforcement would interfere with the drilling of holes along only one rail location. However, if 9-inch holes were drilled, both the first and third layers of No. 18 reinforcement would be cut. Design Change Authorization No. 7041 was written for authorization to cut the uppermost No. 18 bar at only one rail location, but did not reference authorization to cut the lower No. 18 bar. DCA-7041 also stated that the expansion bolts and base plates may be moved in the east-west direction to avoid interference with reinforcement running in the north-south direction. The information, described in DCA-7041, was substantiated by Gibbs & Hill calculations. If the ten holes were actually drilled 9 inches deep, then the allegation that the reinforcement was cut without proper authorization would be valid.

Accordingly, TUEC shall provide:

1. Information to demonstrate that only the No. 18 reinforcing steel in the first layer was cut, or

2. Design calculations to demonstrate that structural integrity is maintained if the No. 18 reinforcing steel on both the first and third layers was cut.

III. Test Programs Area

a. Hot Functional Testing (HFT)

The TRT reviewed a sample of the completed data packages for HFT preoperational test procedures, pertinent startup administrative procedures, NRC inspection reports, and the preoperational test index and its schedule. The TRT also inspected test deficiency reports

(TDRs) that were generated as a result of test deficiencies found prior to and during HFT.

1. Chapter 14 of the FSAR and Regulatory Guide 1.68 provide requirements for the conduct of preoperational testing. In reviewing test data packages, the TRT found that certain test objectives were not met. It appears that the Joint Test Group approved incomplete data packages for at least three preoperational hot functional tests. These were:

Test Procedure	Deficiency
1CP-PT-02-12, "Bus Voltage and Load Survey"	Because acceptable voltages could not be achieved with the specified transformer taps, they were changed. A subsequent engineering evaluation required returning to the original taps, but no retest was performed.
1CP-PT-34-05, "Steam Generator Narrow Range Level Verification"	Level detectors 1-LT-517, 518 and 529 were replaced with temporary equipment of a design that was different from that which was to be eventually installed
1CP-PT-55-05 "Pressurizer Level Control"	Level detector 1-LT-461 appeared to be out of calibration during the test and was replaced after the test. The retest approved by the JTG was a cold calibration rather than a test consistent with the original test objective, which was to obtain satisfactory data under hot conditions.

Accordingly, TUEC shall review all complete preoperational test data packages to ensure there are no other instances where test objectives were not met, or prerequisite conditions were not satisfied. The three items identified by the TRT shall be included, along with appropriate justification, in the test deferral packages presented to the NRC.

2. The TRT noted during a review of HFT completed test data that the JTG did not approve the data until after cooldown from the test. The tests are not considered complete until this approval is obtained. In order to complete the proposed post-fueling, deferred preoperational HFT, the JTG, or a similarly qualified group, must approve the data prior to proceeding to initial criticality. The TRT did not find any document providing assurance that TUEC is committed to do this.

   Accordingly, TUEC shall commit to having a JTG, or similarly qualified group, review and approve all post-fueling preoperational test results prior to declaring the system operable in accordance with the technical specifications.

3. The TRT pointed out that in order to conduct preoperational tests at the necessary temperatures and pressures after fuel load, certain limiting conditions of the proposed technical specifications cannot be met, e.g., all snubbers will not be operable since some will not have been tested.

   Accordingly, TUEC shall evaluate the required plant conditions for the deferred preoperational tests against limiting conditions in the proposed technical specifications and obtain NRC approval where deviations from the technical specifications are necessary.

4. Data for the thermal expansion tests (which have not yet been approved by the JTG) did not provide for traceability between the calibration of the measuring instruments and the monitored locations, as required by Startup Administrative Procedure-7. The information was separately available in a personal log held by Engineering.

   Accordingly, TUEC shall incorporate the information necessary to provide traceability between thermal expansion test monitoring locations and measuring instruments. TUEC shall also establish administrative controls to assure appropriate test and measuring equipment traceability during future testing.

b. Containment Intergrated Leak Rate Testing (CILRT)

The TRT reviewed the data package for the CILRT performed on Unit 1, and discussed the conduct of the test with TUEC and NRC personnel who participated in or witnessed it.

Apparently after repairing leaks found during the first two attempts, the third attempt at a CILRT was successful. It was successfully completed after three electrical penetrations were isolated because the leakage through them could not be stopped. Though the leaks were subsequently repaired and individually tested with satisfactory results, NRC approval was not obtained to perform the CILRT with these penetrations isolated. In addition, leak rate calculations were performed using ANSI/ANS 56.8, which is neither endorsed by the NRC nor in accordance with FSAR commitments.

Accordingly, TUEC shall identify to NRC any other differences in the conduct of the CILRT as a result of using ANSI/ANS 56.8 rather than ANSI N45.4-1972. Additionally, TUEC shall identify to NRC all other deviations from FSAR commitments.

c. Prerequisite Testing

The TRT reviewed FSAR commitments, startup administrative procedures, prerequisite test records, craft personnel qualification records, and discussed them with startup and craft management personnel. The TRT also observed test support craft personnel at work and interviewed some of them to gain familiarity with their attitudes and capabilities.

The review of test records revealed that craft personnel were signing to verify initial conditions for tests in violation of startup Administrative Procedure-21, entitled: "Conduct of Testing" (CP-SAP-21). This procedure requires this function to be performed by System Test Engineers (STE). Startup management had issued a memorandum improperly authorizing craft personnel to perform these verifications on selected tests.

Accordingly, TUEC shall rescind the startup memorandum (STM-83084), which was issued in conflict with CP-SAP-21, and ensure that no other memoranda were issued which are in conflict with approved procedures.

d. Preoperational Testing

The TRT assessed the preoperational test program by reviewing administrative procedures, interviewing startup personnel, and examining test records, schedules, system assignments, subsystem definition packages, and the master data base.

Problems found with test data are addressed in section III.a of this enclosure. The TRT also found that STEs were not being provided with current design information on a routine, controlled basis, and had to update their own material when they considered it appropriate.

Accordingly, TUEC shall establish measures to provide greater assurance that STEs and other responsible personnel are provided with current controlled design documents and change notices.

Attachment 4

Docket Nos.: 50-445
and 50-446

OCT 5 1984

Texas Utilities Electric Company
Attn: M. D. Spence, President, TUGCO
Skyway Tower
400 North Olive Street
Lock Box 81
Dallas, Texas 75201

Dear Mr. Spence:

Subject: September 18, 1984 Letter, D. G. Eisenhut to M. D. Spence,
Re: Comanche Peak Review

During our meeting on September 18, 1984 at Bethesda, Maryland, we discussed the technical issues regarding Comanche Peak which the NRC Technical Review Team identified as having potential safety implications and thus requiring additional information. The subject letter listing these items and the information that we requested were provided to you during that meeting.

We have since discovered some typographical errors in the Enclosure to the September 18, 1984 letter and provided Mr. John Merritt of your staff with a marked-up copy of that letter on September 21, 1984. Enclosed for your information is an errata to the letter.

Sincerely,

Original signed by
Darrell G. Eisenhut

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc w/enclosure:
See next page

Errata
To Enclosure 1 to September 18, 1984 Letter,
D. G. Eisenhut to M. D. Spence

1. Page 2, line 1,

   Panel CP1-ECPRCB-14
   should be
   Panel CP1-ECPRCB-04

2. Page 2, 8th line from bottom of page

   Panel CP1-EC-PRCP-06
   should be
   Panel CP1-EC-PRCB-06

3. Page 4, item 3

   Control Panel CP1-EC-PRCP-06
   should be
   Control Panel CP1-EC-PRCB-06

4. Page 9, 3rd line from bottom of first full paragraph

   Sections 3.4.1.1.1
   should be
   Sections 3.8.1.1.1

5. Page 10, top of page, item 7

   Lowered Suspended Ceiling
   should be
   Louvered Suspended Ceiling

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Attachment 5

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NOV 29 1984

Docket Nos.: 50-445
and 50-446

Mr. M. D. Spence
President
Texas Utilities Generating Company
400 North Olive Street
Lock Box 81
Dallas, Texas 75201

Dear Mr. Spence:

Subject: Comanche Peak Review

On July 9, 1984, the staff began an intensive onsite effort to complete a portion of the reviews necessary for the staff to reach its decision regarding the licensing of Comanche Peak, Unit 1. The onsite effort covered a number of areas, including allegations of improper construction practices at the facility.

On September 18, 1984, the NRC met with you and other Texas Utilities Electric Company representatives to provide you with a number of technical issues in the electrical/instrumentation, civil/structural, and test program areas having potential safety implications. The issues discussed constitute a portion of the technical issues and allegations being evaluated by the Technical Review Team (TRT).

The activities of the TRT have progressed to the point where it is appropriate to provide you with a status of additional items under review and to request additional information. These items, in the coatings, mechanical, and miscellaneous areas, are listed in the enclosure to this letter. Further background information regarding these issues will be published in a Supplement to a Safety Evaluation Report (SSER), which will document the TRT's overall assessment of the significance of the issues examined.

The items in the enclosure to this letter cover only a portion of the TRT's effort. The TRT's ongoing evaluation, QA/QC review and conversations with allegers may reveal additional items in the coatings, mechanical, and miscellaneous areas for which additional requests for information may be appropriate. Also, the TRT evaluation of QA/QC issues, and its consideration of the programmatic implications of these findings, are still in progress. A summary of these issues will be provided to you at a later date.

You are requested to submit additional information to the NRC, in writing, including a program and schedule for completing a detailed and thorough assessment of the issues identified in the enclosure to this letter. This program plan and its implementation will be evaluated by the staff before NRC considers the issuance of an operating license for Comanche Peak, Unit 1. The program plan should address the root cause of each problem identified and its generic implications on safety-related systems, programs, or areas. You should also address the collective significance of these deficiencies. Your program plan should also include the proposed TUEC action to assure that such problems will not occur in the future.

This request is submitted to you in keeping with the NRC practice of promptly notifying applicants of outstanding information needs that could potentially affect the safe operation of their plant. Future requests for additional information of this nature will be made, if necessary, as the activities of the TRT progress.

Sincerely,

Frank J. Miraglia
for
Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure: As stated
cc w/enclosure:
See next page

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REQUEST FOR ADDITIONAL INFORMATION

IV. Protective Coatings Area

a. Surveillance and Test Program for Coatings

The protective coatings Technical Review Team (TRT) reviewed the backfit program, design basis accident qualifications, traceability, application and repair procedures, training, coating exempt log and dispositioning of non-conformance reports. Concurrently, the staff is evaluating the effects on containment emergency sump performance of paint and insulation debris. The results of the two concurrent reviews will be combined in one supplemental safety evaluation which is scheduled to be issued by January 1985. Actions required for resolution of protective coatings issues will be delineated in the supplement.

V. Mechanical Area

a. Inspection for Certain Types of Skewed Welds in NF Supports

The TRT investigated inspection procedures of Brown & Root (B&R) for welds in pipe supports designed to ASME III Code, Subsection NF. The TRT found that no fillet weld inspection criteria existed for certain types of skewed welds. By definition, skewed welds are those welds joining (1) two non-perpendicular or non-colinear structural members, or (2) two members with curved surfaces or curved cross sections, such as a pipe stanchion (a section of pipe used as a structural member) welded to another pipe stanchion or to a curved pipe pad. Notice that for type (2), the effect of curvature at the weld connection induces skewed considerations, even though the two joining members are physically perpendicular. The B&R weld inspection procedures CP-QAP-12.1 and QI-QAP-11.1-28 for NF supports have addressed type (1) skewed welds; however, the TRT found that QI-QAP-11.1-28 did not include weld inspection criteria for type (2) skewed welds. Although the TRT was told by B&R personnel that procedure QI-QAP-11.1-26 for piping weld inspection was used, since such weld connections were similar in configuration to a pressure boundary stanchion attachment weld, no evidence documenting the use of this inspection procedure was provided to the TRT. According to records reviewed by the TRT, these welds were actually categorized as "all other welds" rather than "skewed welds" on the required QC checklist. Instead of using fillet weld gauges for measuring the size of nonskewed welds, welders were supposed to use a straight edge and a steel scale for measurement of a type (2) skewed weld, as described in QI-QAP-11.1-28. In addition, due to the variable profile along its curved weld connection, the weld size should have been measured at several different locations. The lack of inspection criteria and lack of verification of proper inspection procedures being conducted for type (2) skewed welds are a violation of ASME Code for NF supports committed to by TUEC in FSAR Section 5.2.1 and a violation of Criterion XVII in Appendix B of 10 CFR 50.

The TRT reviewed weld inspection procedures, weld data cards, and visually inspected several type (2) skewed welds in randomly sampled NF supports where pipe stanchions were used. Although the small sample of welds inspected by the TRT are acceptable, due to deficiencies in inspection records and the apparent lack of inspection criteria, the TRT is not certain whether other type (2) skewed welds were inspected properly. This is a generic issue involving many NF supports in various safety-related systems. The lack of documented inspections and criteria for type (2) skewed welds in NF supports represents a safety concern regarding the possible existence of under-sized welds in supports which are required to resist various design loads.

Accordingly, TUEC shall

(1) Revise B&R weld inspection procedures CP-QAP-21.1 and QI-QAP-11.1-28 to properly address type (2) skewed welds of stanchion to stanchion and stanchion to pipe pad; and,

(2) provide evidence to verify that previous inspections of these types of skewed welds were performed to the appropriate procedures.

b. <u>Improper Shortening of Anchor Bolts in Steam Generator Upper Lateral Supports</u>

The TRT was informed that some anchor bolts in the steam generator upper support beams were shortened during installation to less than the length shown on the design drawing without proper authorization. The TRT was told that the bolt cutting incident occurred either because the hole of the anchor device was filled with debris, or the threaded portion of the bolt had concrete mix stuck to it. There are 18 bolts at each end of each of 4 beams, totalling 144 bolts. There is one beam for each steam generator. The bolt threads into an anchor device embedded in the concrete wall. The acceptable bolt length or the length of bolt available for threading into the anchor device is vital to ensure structural capability of the support beams.

The TRT attempted to review TUEC records for ultrasonic (UT) measurement results and general installation practices. The TRT was told that ultrasonic testing of these types of bolts was not a procedural requirement; however, TUEC was unable to provide any other installation records for TRT review. The TRT concludes that such unauthorized bolt cutting and lack of installation inspection records is a violation of QA procedures and Criterion XVII in Appendix B of 10 CFR 50. Since the support beams are essential to provide lateral restraint for the steam generator during a LOCA or seismic event, adequate anchoring capability of the bolts has safety significance and, as a result, appropriate measures are needed to ensure conformance with General Design Criterion 1 of 10 CFR 50.

Accordingly, TUEC shall provide evidence, such as ultrasonic measurement results, to verify acceptable bolt length. Should unauthorized bolt cutting be verified, TUEC shall:

(1) replace shortened bolts with bolts of proper length, or provide analysis to justify the adequacy of shortened bolts as installed; and

(2) provide justification or propose measures to ensure that no similar concern exists for bolting.

c. Design Consideration for Piping Systems Between Seismic Category I and Non-Seismic Category I Buildings

In April 1984 the Comanche Peak Special Review Team (SRT), formed and coordinated between NRR, IE and Region II and IV, performed a limited review of Comanche Peak. The TRT, in reviewing the SRT findings in the area of piping design considerations, has discovered that piping systems, such as Main Steam, Auxiliary Steam and Feedwater, are routed from the Electrical Control Building (seismic category I) to the Turbine Building (non-seismic category I) without any isolation. To be acceptable, each seismic category I piping system should be isolated from any non-seismic category I piping system by separation, barrier or constraint.

If isolation is not feasible, then the effect on the seismic category I piping of the failure in the non-seismic category I piping must be considered (CPSES FSAR 3.7B.3-13.1).

For CPSES, FSAR section 3.7B.2.8 establishes that the Turbine Building is a non-seismic category I structure and failure is postulated during the seismic (SSE) event. The effect of Turbine Building failure on any non-isolated piping routed through the Turbine Building from any seismic category I building must be considered.

In addition, for non-seismic category I piping connected to Seismic Category I piping, the dynamic effects of the non-seismic category I piping must be considered in the seismic design of the seismic category I piping and supports, unless TUEC can show that the dynamic effects of the non-seismic category I piping are isolated by anchors or restraints. The anchors or restraints used for isolation purposes must be designed to withstand the combined loading imposed by both the seismic category I and non-seismic category I piping.

Accordingly, TUEC shall provide analysis and documentation that the piping systems routed from seismic category I to non-seismic category I buildings meet the stated FSAR criteria.

d. Plug Welds

The TRT investigated alleged generic problems regarding uncontrolled repairs to holes existing in pipe supports, cable tray supports and base plates in Units 1 and 2. These holes, which had been misdrilled during fabrication, were repaired by plug welds. Since these supports are Seismic

Category I supports and the effects of the welds have not been evaluated, this constitutes a violation of Criteria IX and XVI of Appendix B to 10 CFR 50. Region IV inspections have confirmed the existence of such welds in cable tray supports located in the Unit 2 Cable Spreading Room.

Although the effects of unauthorized, undocumented and uninspected plug welds in some locations (e.g., the webs of I-beams or in structural members in compression) will be inconsequential, their effects in critical locations (e.g., flanges of I-beams in flexure or in structural members in tension) in critically loaded supports or base plates could affect their structural integrity and intended function.

Accordingly, TUEC shall perform one of the following:

(1) Modify its proposed plan to Region IV (TXX-4183 and TXX-4259) to include a sampling inspection of all areas of the plant having plug welds, to include cable tray supports, pipe supports and base plates. Propose alternate methods of inspection where the oblique lighting method is not viable (e.g., locations covered by heavy coats of paint). Perform an assessment of the effects on quality due to uncontrolled plug welds found during the proposed inspection, as modified above. Submit a report documenting the results of the inspection and assessment to the NRC for review.

(2) Perform bounding analyses to assess the generic effects of uncontrolled plug welds on the ability of pipe supports, cable tray supports and base plates to serve their intended function. Submit a report documenting the results of the assessment to the NRC for review.

e. <u>Installation of Main Steam Pipes</u>

The TRT investigated an allegation that a Unit 1 main steam line had been installed incorrectly and had been forced into proper alignment after flushing operations by use of the main polar crane and come-alongs. It was also claimed that pipe supports had been modified to maintain the line in its forced position and vibrations following detachment of the flushing line could have damaged the main steam line. Based on its investigation, the TRT determined that the alleged incident pertained to restoration of the Unit 1, loop 1 main steam line to its initial, correct installation position. (The line had shifted during flushing operations due to the weight of the added water and because the temporary supports sagged.) The TRT also determined that the modifications to permanent pipe supports were necessary to provide proper support to the main steam line in its restored position (initial designs for and construction of the supports had been based on the shifted position of the line) and, although the alleged vibrations could not be confirmed, their associated stresses might not have damaged the main steam line. (The highest stresses would have occurred in the weaker, temporary flushing line.) The TRT review of a TUEC analysis, performed 1 year after the incident, concluded that the analysis was incomplete. An evaluation for the full sequence of events leading up to the incident had not been performed. The TRT review of Gibbs & Hill Specification No. 2323-MS-100 indicated that there were inadequate requirements and construction practices for the support of the main steam line during flushing, and for temporary supports for piping and equipment in general. In particular, evaluations to assure the adequacy of temporary supports during flushing and installation were not required. The deficiencies in the analyses, specifications and construction practice identified above constitute a violation of Criterion V of Appendix B to 10 CFR 50.

Accordingly, TUEC shall:

(1) Modify Gibbs & Hill Specification No. 2323-MS-100, and institute procedures for support of the main steam line during flushing and for temporary supports for piping and equipment in general to assure that the quality of piping and equipment are not affected.

(2) Perform an assessment of stresses in the portions of the Unit 1, loop 1, main steam and feedwater lines that were affected in the sequence of events involved during their initial installation, flushing and final installation. Conditions requiring stress analysis are:

    (a) Flushing condition when the lines were full of water and temporary supports had sagged or settled.

    (b) Disconnecting condition when vibrations of the temporary line could have occurred.

    (c) Lifting condition when forces were applied by the polar crane and come-alongs.

    These assessments shall be based on appropriate piping configurations involved.

(3) Perform a non-destructive examination of locations in the Unit 1, loop 1, main steam and feedwater piping where stresses were exceeded during the conditions of concern in a. through c. above.

(4) Review the existing baseline UT examinations for those portions of the Unit 1, loop 1, main steam and feedwater involved in all the conditions of concern in a. through c., above, for unacceptable indications.

(5) Review records of hydrostatic testing of the main steam and feedwater line to verify the quality of piping involved in the incident.

(6) Provide similar assessments for circumstances involved in a lifting incident identified during the TRT inspection for the Unit 1, loop 4, main steam line.

(7) Provide assessments of effects on quality of safety-related piping and equipment which were involved in similar incidents of sagging, settlements and failures, if any, of temporary supports.

(8) Submit the results of analyses, examinations and reviews in a documented report for NRC review.

VI. Miscellaneous Area

a. Gap Between Reactor Pressure Vessel Reflective Insulation (RPVRI) and the Biological Shield Wall

The TRT investigated an allegation that the Unit 1 reactor pressure vessel outer wall was touching the concrete biological shield wall. A TRT review of existing documentation and discussions with TUEC personnel indicated that this allegation was not factual. However, a significant construction deficiency report, submitted pursuant to 10 CFR Part 50.55(e), on August 25, 1983, documented that unacceptable cooling occurred in the annulus between the RPVRI and the shield wall during hot functional testing, apparently because of the existence of an inadequately sized annulus gap and possibly because the presence of construction debris in the annulus. TUEC corrected the situation by modifications to allow increased air flow for proper heat dissipation and by removal of the construction debris. TUEC representatives indicated that testing to verify the adequacy of the cooling flow will take place when additional hot functional testing is conducted. Information gathered by the TRT during the investigation indicated that a design change in the RPVRI support ring (i.e., locating the ring outside rather than inside the insulation) resulted in a limited clearance between the RPVRI and the shield wall. The TRT review of the 50.55(e) report revealed that TUEC failed to: (1) address the fundamental issue of the design change impact on annulus cooling flow, and (2) determine whether Unit 2 was similarly affected.

Accordingly, TUEC shall:

(1) Review their procedures for approval of design changes to non-nuclear safety-related equipment, such as the RPVRI, and make revisions as necessary to assure that such design changes do not adversely affect safety-related systems.

(2) Review procedures for reporting significant design and construction deficiencies, pursuant to 10 CFR Part 50.55(e), and make changes as necessary to assure that complete evaluations are conducted.

(3) Provide an analysis which verifies that the cooling flow in the annulus between the RPVRI and the shield wall of Unit 2 is adequate for the as-built condition.

(4) Finally, verify during future Unit 1 hot functional testing that completed modifications to the RPVRI support ring now allow adequate cooling air flow.

The TRT noted that control of debris in critical spaces between components and/or structures was identified as an issue, both in the investigation of this allegation and the civil/structural area item II.c (Maintenance of Air Gap Between Concrete Structures), contained in Darrell G. Eisenhut's September 18, 1984, letter to TUEC. Accordingly, TUEC shall also:

(1) Identify areas in the plant having critical spacing between components and/or structures that are necessary for proper functioning of safety-related components, systems or structures in which unwanted debris may collect and be undetected or be difficult to remove;

(2) Prior to fuel load, inspect the areas and spaces identified and remove debris; and,

(3) Subsequent to fuel load, institute a program to minimize the collection of debris in critical spaces and periodically inspect these spaces and remove any debris which may be present.

b. Polar Crane Shimming

The TRT investigated the installation of the polar crane rail support system by visual inspection, review of associated documentation, and discussions with TUEC representatives and their contractors. Region IV Inspection Report 50-445/84-08; 50-446/84-04 and Notice of Violation, dated July 26, 1984, documented that gaps on the Unit 1 polar crane bracket and seismic connections exceeded design requirements. In Texas Utilities Generating Company responses of August 23, 1984, and September 7, 1984, the gaps were attributed to crane and bolting self-adjustment resulting from crane operation. A site design change (DCA-9872, Revision 4, dated August 24, 1984) was issued to document the acceptability of the gaps in excess of 1/16 inch which were identified in the above NRC inspection report.

During further investigation of the allegation that shims for the rail support system of the polar crane had been altered during installation, the TRT observed gaps which may have been excessive between the crane girder and the girder support bracket. Detailed specifications addressing the gap tolerance in the girder seat connections did not exist; however, Gibbs & Hill letter GHF-2207, dated November 28, 1977, stated that the "seated connections will not require shimming since the area in bearing is at least the width of the bottom flange of the crane girder." Contrary to this Gibbs & Hill assumption, the TRT observed nine girders with gaps which extended under the bottom flange that reduced the bearing surface to less than the 20-inch flange width stated in the letter. The TRT also observed conditions which indicated that the crane rail may still be moving in a circumferential direction, that three rail-to-rail ground wires were broken, that two shims have partially worked out from under the rail, and that two cadwelds were broken.

Accordingly, TUEC shall:

1. Inspect the polar crane rail girder seat connections for the presence of gaps which reduce the bearing surface to less than the width of the bottom flange and perform an analysis which will determine whether existing gaps are acceptable or require corrective action.

2. Determine if additional rail movement is occurring and, if so, provide an evaluation of safety significance and the need for corrective action.

3. Perform a general inspection of the polar crane rail and rail support system, correct identified deficiencies of safety significance, and provide an assessment of the adequacy of existing maintenance and surveillance programs.

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